NUCLEAR SAFETY

A REVIEW OF RECENT DEVELOPMENTS

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foreword

This quarterly journal is a Technical Progress Review prepared by the Oak Ridge National Laboratory at the request of the Office of Technical Information, U. S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. *Nuclear Safety* is not a comprehensive abstract of all literature published in this field during a given quarter; rather it is a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered to be in the province of health physics. Even with these exclusions, nuclear safety overlaps such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

Three distinctly different types of articles are in this issue of *Nuclear Safety:* reviews of current literature, special review articles on specific topics, and comparative studies of various aspects of U. S. power reactors. The editors feel that each of these articles makes a necessary and distinctive contribution to this journal. The special review articles permit discussion of pertinent subjects which cannot be adequately considered by reference to only the current literature. The comparative studies direct attention to acceptable practice in the U. S. power reactor industry. The current review articles, however, constitute the major portion of this issue.

All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. Interpretations, in any article, represent the opinions of the editors, who are employed by the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information.

In addition to the invited contributions, many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

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Section I

NUCLEAR SAFETY CRITERIA

Reactor Inspection

By PETER A. MORRIS*

Until Aug. 1, 1960, inspection of licensed reactors, and those being built under a construction permit, was carried out by the U. S. Atomic Energy Commission (AEC) Division of Inspection. This Division, which was created specifically by the Atomic Energy Act of 1954,[1] is charged with "gathering information to show whether or not the contractors, licensees, and officers and employees of the Commission are complying with the provisions of this Act (except those provisions for which the Federal Bureau of Investigation is responsible) and the appropriate rules and regulations of the Commission."[2,3] On Aug. 1, 1960, the responsibility for inspection of licensed reactors, and those with construction permits, was transferred to the newly formed Division of Compliance.[2] At the present time, the Division of Compliance carries out the actual field inspections of licensed facilities, and the Division of Inspection retains its over-all responsibility for ascertaining that adequate inspection activities are being carried out for licensed reactors, as well as for AEC-owned and contractor-operated reactors.

Both the Division of Licensing and Regulation and the Division of Compliance report to the Assistant General Manager for Regulation and Safety.[2] These Divisions, with the advice of the Advisory Committee on Reactor Safeguards (ACRS), carry out the AEC's responsibility for protection of the health and safety of the public with regard to the operation of licensed nuclear reactors.[1,3]

Organization for Inspection

The Division of Compliance, established Nov. 13, 1959, was staffed originally on June 13, 1960, and acquired personnel for reactor inspection work on Aug. 1, 1960. Within the Division of Compliance are a Materials Inspection Branch and a Reactor Inspection Branch. Responsibility for inspection of the facilities of the approximately 10,000 materials-license holders (source, by-product, and special nuclear materials) has been delegated to eight of the Commission's field offices. Responsibility for inspection of privately owned reactor facilities (approximately 50 operating; another 40 under construction or planned) remains with the headquarters staff. The Assistant Director for Reactors, Division of Compliance, and three inspection specialists, who are located at field offices, directly inspect all facilities operating under license or being built under a construction permit. Occasionally, technical consultants also assist in inspections. The inspectors work closely with the Division of Licensing and Regulation to achieve effective communication of field findings. Formal recommendations of the Division are made to the Assistant General Manager for Regulation and Safety of the AEC,

*Dr. Peter A. Morris, Assistant Director for Reactors, Division of Compliance, AEC, was associated with the AEC Headquarters Division of Inspection from October 1957 to July 1960. From 1951 until 1957, Dr. Morris was employed by E. I. du Pont de Nemours & Co., Inc., at the Argonne National Laboratory, at the Du Pont home office in Wilmington, Del., and at the Savannah River Laboratory. From 1946 to 1948, he was with the Bartol Research Foundation. He received the B.A. degree in mathematics from Swarthmore College in 1943 and the Ph.D. degree in physics from the University of Virginia in 1951. He served in the U. S. Air Force from 1943 to 1946. Dr. Morris is a member of Sigma Xi and Phi Beta Kappa honorary societies and of the American Nuclear Society.

and enforcement or other action is taken through the Division of Licensing and Regulation.

Inspection Policy

The policy for the reactor inspection program has been and is to use inspectors who have had extensive experience in both the design and operation of reactors. The emphasis has been to concentrate the efforts and attention of these men on those aspects of reactor design and operation which are related to the health and safety of the public. The Division of Compliance does not conduct the more or less conventional types of inspection, such as inspection for compliance with electrical codes, pressure-vessel codes, and building codes. The techniques and bases for these inspections have been established and agreed upon by national organizations of considerable stature, and therefore the Division of Compliance participates only to the extent of assuring itself that such inspections are made responsibly and independently by specified persons or groups. The features and equipment peculiar to nuclear reactors, however, and directly related to safety of operation may be inspected in detail; the items might include a new control-system design, initial pressure test of a containment vessel, or calibration of certain instrumentation, for example.

Safety of nuclear facilities is the responsibility of licensee management, and this basic responsibility cannot be delegated. For this reason, in so far as practicable, initial contacts by reactor inspectors are made with management personnel to explain and describe the functions, responsibilities, and authority of the Division of Compliance. At the conclusion of all visits, it is also desired that management be informed of the findings and conclusions of the inspector, recognizing that these are subject to review within the Commission. It also follows, for privately owned reactors, that basic responsibility of licensee management for safe reactor operation cannot be shared by the Division of Compliance or others in the AEC.

Because of the unique opportunities of the inspectors to visit many facilities and thus to become acquainted with a broad range of operating experience, an obligation is felt to disseminate information and to review experiences as much as possible, with due regard for proprietary information and unambiguous identification of personal opinions and predilections.

The Division of Compliance does not inspect Commission-owned reactors. Responsibility for inspection of such reactors is that of the managers of the operations offices having jurisdiction over them.[2] The Division of Inspection does have responsibility, however, as stated in the Act,[1] to gather information to show whether the managers and contractors are carrying out their respective responsibilities.

Methods

The scope and methods of reactor inspection vary widely, depending on the type of reactor, the state of construction or operation, the experience and ability of the operator, etc. For a reasonably large reactor, or a reactor of new design, the inspection function involves the following phases:

1. Familiarization of inspection personnel with the facility during design review and hazards evaluation activities.
2. Observation and review of construction progress.
3. Observation and review of equipment and procedures before nuclear operation.
4. Observation and review of the initial loading of fuel, initial reactor startup, and operational program.
5. Periodic observation and review during normal operation.

Inspection activities may include visual inspections; observation of acceptance or performance tests; observation of simulated or actual operation; review of organization, administrative, and operating procedures; review of operating records, logs, and reports; and discussion with operating, technical, and managerial personnel. The frequency of inspection visits varies widely. Routine visits to an "off-the-shelf" training reactor may be spaced a year or more apart. Visits to a power reactor during the final acceptance testing preceding initial nuclear operation may be as frequent as several days per week.

Experience

Inspection experience has been generally favorable in that no gross violations of the rules and regulations have appeared and no cases of complete inadequacy of equipment, procedures,

or people have been found. A number of cases have been noted in which operational safety could have been, and was, improved distinctly by relatively minor changes or additions. (A clear-cut example of this type of situation is proper location of neutron source and detector with respect to the core during loading and startup operations.) In several cases it has been found that the owners of training or research reactors were relatively inexperienced. In these cases the presence of consultants or the reactor manufacturer's personnel was required until the owner acquired adequate proficiency.

Because of the preliminary status of standards and codes in the reactor field, any appraisal of operational safety necessarily involves judgment on the part of the appraiser. As in other fields, the degree of conservatism in the reactor business appears to be a function of experience. In inexperienced groups there have sometimes been strong pressures, because of scientific enthusiasm and business competition, opposing a prudent regard for sound safety practices. Experienced groups have learned that any compromises of conservative safety attitudes or practices in reactor operation invariably lead to malfunction or mishap.

Inspections in Other Countries

Implementation of regulatory requirements for reactors appears to be further advanced in the United States than in any other country. The Canadian Atomic Energy Control Act[4] permits the Atomic Energy Control Board to make regulations, including those regarding the licensing of facilities and the use of atomic energy. Although such regulations are in the process of revision, it is understood that inspection of Canada's reactors will be done, on an informal basis only, by members or representatives of the Board, at least until the number of reactors in Canada becomes large enough to justify a full-time inspection activity.

In Japan, licensed reactors are allowed to be put into operation only after inspection and examination, by direction of the Prime Minister.[5] Details of this mechanism are not available.

In England, government reactors are reviewed and inspected by the Safeguards Division of the Atomic Energy Authority Health and Safety Branch.[6] In the future, nongovernment reactors are to be licensed and inspected by the Ministry of Power. This program is currently being formulated.

Information is not available at present concerning inspection activities in Russia or France.

Future AEC Inspection

The Division of Compliance exercises the direct inspection function of the AEC for licensed reactors.[2] Eventually, responsibility for inspection of certain classes of reactors, or specific reactors, will be delegated to some of the operations offices, under the guidance of policies and procedures developed by the Headquarters Division. (Peter A. Morris)

Shipment of Spent Fuel Elements

Processing of spent fuel from power reactors will generally involve the transportation of a considerable tonnage of radioactive material over hundreds of miles. For the present, the processing will be done in government-owned plants.[7] Later, as the number of reactors increases, it is expected that privately owned processing plants will be built. The locations of the government plants and the power reactors they are now expected to serve[8] are shown in Fig. 1. The fuel-processing site has not yet been designated for a number of the proposed power reactors.

Irradiated fuel is transported in heavily shielded containers called casks. From a hazards standpoint, a loaded spent-fuel cask is in some respects like a portable nuclear reactor, and important safety requirements in handling casks are therefore the prevention of criticality, provision for adequate heat removal, adherence to structural integrity, etc. If the fuel in the shipping cask should "go critical," the cask would become, at the very least, a hazardous source of neutrons and radioactive materials. If a criticality incident resulted from a wreck that also caused a rupture of the cask, the incident could lead to dangerous dispersal of fission products. Rupture or meltdown of a spent fuel element during transport could also lead to the release of radioactive particulate matter and volatile fission products. Thus an inadequately designed or handled cask would be a hazard to anyone in its vicinity.

In order to preclude or minimize the hazards, the AEC is preparing regulations govern-

Figure 1 — Map showing locations of government-owned fuel-processing plants and the power reactors the plants will serve.

ing the shipment of irradiated solid fuel elements. A proposed set of regulations has been published[9] for comment by interested persons and organizations. On the assumption that the revisions based on the comments will be minor, it is believed that a discussion of the regulations in their present form, with emphasis on the apparent intent rather than on the precise wording used, is timely.

The regulations apply only to the shipment of solid fuel elements. Presumably, regulations relative to the shipment of liquid fuels will be issued when the need arises.

The regulations state that a license to transport fuel must be obtained from the AEC. Issuance of a license is considered on the basis of information submitted by the applicant, including engineering analyses of criticality, heat transfer, radiation shielding, structural integrity, control of contamination of shipping casks, and the handling procedures to be employed.

In addition to the regulations governing shipment, general criteria for acceptance of casks by the processing plants have been formulated for the guidance of shippers.[10] Also, the individual processing plants have prepared their own acceptance criteria.[11–13] It should be pointed out that these criteria are generally for routine operations, and, as such, there would be no additional charges. Special handling of casks (if they can be handled safely) would be possibly approved, provided the shipper pays for all additional costs.

Criticality Control

The shipper may employ either mass control or geometry control to prevent criticality. If mass control is employed, no more than 75 per cent of a critical mass may be placed in a single cask. In determining critical mass, credit may be taken for poisons built into the cask or fuel elements. In all other respects, however, the worst credible set of conditions must be assumed. Fuel must be taken at its maximum reactivity; i.e., it must be treated as unirradiated fuel or, if it contains built-in poisons, as fuel irradiated to the point of greatest reactivity. The presence of the amount of hydrogenous material giving maximum reactivity must be assumed. Similarly, it must be assumed that the spacing between fuel elements is that which gives maximum reactivity, or it must be shown that the fuel elements cannot be

arranged into a configuration more reactive than that for which the shipment is designed. Unless it can be established that meltdown of the fuel elements is impossible, the fuel must be considered as melted fuel in the most reactive array. Where geometry control is employed, each controlling dimension must incorporate an allowance for defined uncertainties in experiment or calculation and, in addition, a 10 per cent safety factor.

The applicant for a license must demonstrate that shipments involving more than one cask will be safe from criticality based on interactions between casks. In the present form of the proposed regulations, no provision is made regarding the possibility of interaction between casks from different shippers that might pass near one another, as could easily occur at or near a processing site. This situation has been called to the Commission's attention[14] by the Chairman of ANS Standards Project 8.

Heat Removal

The use of either a liquid or a gaseous coolant is permitted by the regulations. However, the use of liquids other than water is generally not acceptable to the processing plants when such liquids could contaminate storage-basin water.[10] It is likely that water will be used almost universally as the coolant, since it is planned that the loading of spent fuel into casks will be carried out under water at all U. S. power-reactor stations. For a spent-fuel load that liberates very little heat, it may be necessary to use an antifreeze in water. In that event, special arrangements would have to be made with the processing plant. If a liquid coolant is used, its maximum temperature must be restricted to 20°F below its boiling point at the highest elevation on the route. The temperature of the accessible surfaces of the cask must be limited to 180°F.

Coolant that is in contact with the fuel may not be circulated outside the cask shielding. A secondary coolant is required if heat sinks in the form of radiators separate from the cask are to be used. This requirement, together with another which states that the cooling system shall operate at approximately atmospheric pressure, probably will have the effect of limiting the use of gaseous coolants to casks holding small amounts of fuel irradiated to a low level.

Piper and Langhaar[15] have described the design and construction of a 70-ton spent-fuel cask having a finned surface for dissipation of heat. The finned surface provides an excellent solution to the heat-dissipation problem. For highly irradiated fuels, however, its use is limited to loads of the order of 1 ton or less of uranium. This load limit varies widely, of course, depending on irradiation conditions, cooling time, and shape and size of the cask.

An important part of the regulations pertaining to heat removal concerns the possible loss of coolant. The regulations acknowledge the inability to preclude the possibility of the loss of coolant, and they establish criteria intended to minimize the damage that might result. The proposed regulation appears to require that, in the event coolant is lost, no constituent in the cask, including fuel, cladding, and built-in poison, will reach a temperature greater than 180°F below its melting point. Compliance with this regulation would prevent criticality incidents that might otherwise occur if, for example, poison-containing spacers or fuel cladding should melt. This regulation also seeks to prevent the spread of gaseous and volatile fission products which might occur as the result of a fuel meltdown without accompanying criticality.

Prediction of temperatures in the cask may be difficult without some experimentation. Suddath[16] has reported a test in which a simulated Sodium Reactor Experiment (SRE) fuel element positioned in an SRE fuel cask was heated electrically, equilibrium temperatures were measured, and heat-transfer coefficients were calculated on the basis of the measured temperatures. In another set of tests described by Wett,[17] irradiated Oak Ridge Research Reactor (ORR) fuel elements were cooled in stagnant air. It was found that a decay period of 19 hr is sufficient to keep the surface temperature of an ORR fuel element, in air, below a dangerous level. Wett states that, although the tests are not directly applicable to other than ORR fuel elements, the results of the investigation can be used in establishing orders of magnitude estimates for other fuel elements. It is likely that tests similar to these will be needed for each fuel and fuel-cask combination in order to establish compliance with the loss-of-coolant criterion.

Structural Integrity

The irradiated-fuel shipping regulations do not require that a cask maintain its structural integrity under all conceivable accidents. The regulations specify that it shall be assumed that a cover edge of the loaded cask traveling at 44 ft/sec (30-ft drop) will come in contact with a solid object. It must be shown that in this situation the cask will remain covered and intact and the fuel will not be rearranged into a more reactive configuration. It is not required that there be no loss of coolant; the regulations on heat removal cover that situation.

The fuel cask must be free of protruding parts except for handling trunnions, cask lid-lifting outriggers, or cooling fins. Items such as sampling and relief valves and expansion tanks must be recessed or protected from damage by steel guards. The processing plants desire that the bottom surface of the cask cover be free of projections so that it can be placed on the bottom of the fuel storage pool during unloading.

External Radiation Levels

The gamma-radiation level at any readily accessible surface of the cask may not exceed 200 mr/hr. For truck shipments, a maximum level of 10 mr/hr at 1 meter also applies. For rail shipments, the level may not exceed 10 mr/hr at 3 meters.

In commenting on the proposed regulations, the New York Central System[18] urged that additional criteria be imposed for radiation levels at railway car surfaces to protect railway personnel. They also pointed out that the cask labels required under the regulations are of no value when the cask is in a closed car and suggested that external warning signs also be required.

External Contamination Limits

Surfaces of the cask that may become contaminated are to be made of series 300 stainless steel or some similar material that can be decontaminated readily by scrubbing with detergent and water. It is noted that carbon steel, coated with a suitable paint, can be decontaminated easily. However, one drawback in utilizing shipping casks with painted surfaces is the possible acceptance of active surfaces, non-decontaminated by normal techniques, which, upon subsequent flaking of the paint, would be exposed.

Lead is frequently used for cask shielding. For heat transfer, a good bond is required between the lead and the surface material. In the studies described by Piper and Langhaar,[15] it was found that lead bonded better to carbon steel than to stainless steel, and therefore stainless-steel-clad carbon steel was used in the Savannah River cask design.

The regulations provide that a cask shall not be shipped if the surface beta-gamma activity exceeds 4000 dis/min per 100 cm^2 or if the alpha activity exceeds 500 dis/min per 100 cm^2. If a cask were shipped with contamination approaching the limits set in the regulations, additional decontamination would be required before it would be acceptable to the processor at most sites.

Contamination of shipping casks and fuel storage-canal water by ruptured fuel elements has also been discussed in a review by King.[19]

Other Requirements

The cask must be provided with means for sampling the coolant. Samples are taken at the processing plant to determine activity in the coolant. Normal acceptance criteria limit the activity level of such samples as something less than 5×10^{-8} curies/cm^3. Any indication of activity level approaching this value might well indicate the existence of a ruptured fuel element.[11]

The regulations specify that, where there is a pressure-relief valve, a filter shall be provided to prevent, as far as possible, the release of particulate matter.

In addition to the technical and engineering criteria discussed above, the regulations specify requirements for labeling and describe the records that must be kept for the fuel. The Commission must be notified of any accidental criticality, major damage to the shipment, dispersal of fission products, or loss.

Possible Effects of the Regulations

Economic considerations, alone, generally lead to the design of the largest possible spent-fuel cask. It is believed that the proposed regulations, if adopted in substantially their present form, will have the effect of scaling down the size of the casks that might have been designed in the absence of the regulations. The

requirements regarding criticality and heat removal will have a major influence on the cask sizes adopted.

For the present, a conservative approach is the only one that can safely be taken in shipping spent fuel. The regulations appear to be conservative enough to avoid serious accidents. Should experience warrant it, the regulations can, of course, be relaxed in the future.

(W. L. Albrecht)

Handling of Radiation Incidents

There are many facets to the problems encountered in the prevention and handling of nuclear or radiation accidents.[20] It is therefore best that a nuclear safety program have its origin at the time of site selection and extend through the various phases of design, into the selection of processes and equipment, and into the selection and training of personnel.[21] Ideally, such a program should prevent accidents, but, in spite of careful planning, it must be accepted that accidents may eventually occur.[22–26] A plan for handling nuclear and radiation incidents is described here. Installations are required to report any nuclear or radiation incidents to the AEC, and regulations have recently been established to ensure adequate and uniform reporting, as well as followup action to prevent recurrences.

An Emergency Plan

In order to handle a nuclear incident or a radiation accident successfully, an installation must have a well-organized emergency plan and personnel trained to cope with the situation. Failure to plan and rehearse adequately may convert a small incident into a disaster.

Basically, a radiation emergency plan is similar to any nonnuclear emergency plan since its prime purpose is personnel safety and the minimization of property damage. Radiation emergency planning does differ, however, in the following respects:

1. A nonconventional detection and warning system is required.

2. A large variety of specialists, equipment, and facilities must be available or quickly assembled to assist in handling the emergency.[27]

3. There is a need to detain personnel after an evacuation in order to monitor them for radioactivity.

4. The determination of injury to personnel and extent of property damage takes more time and technical effort than in an ordinary accident.

5. Prompt news releases for public understanding are usually helpful because of inordinate fear of radiation by the public.

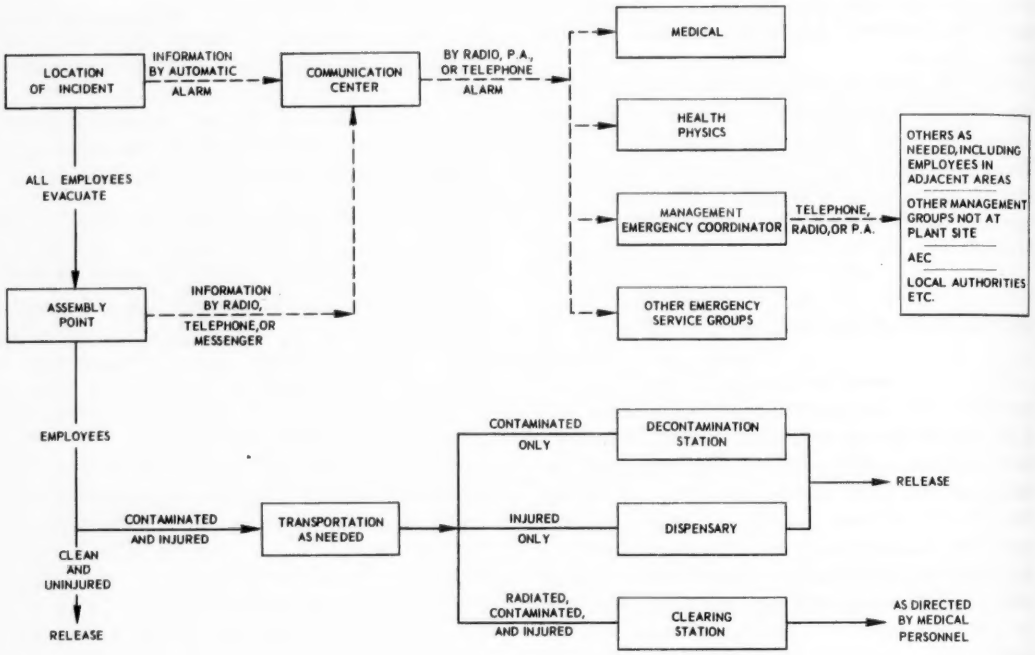
Radiation emergency planning should start with a survey of the installation. All sources of radioactivity and fissionable material should be listed and studied, and the types of potential accidents should be determined, i.e., criticality, high radiation fields, and contamination. An estimate should also be made of the area and number of people which would be involved in the event of the postulated accident.

Once this information is available, the responsibilities for adequate emergency planning and action must be fixed. In general, it is logical to assign these responsibilities to the organization operating the facility that contains the potentially dangerous source, i.e., a reactor, a chemical process, a laboratory, or an office building. The responsible organization must study the potential emergencies within its area, make plans to avoid the occurrence of such emergencies, and make adequate preparations for taking prompt action if an accident occurs. Each organization must ensure that:

1. All personnel are familiar with the building and installation evacuation plans and alarms and the procedure for properly reporting an emergency, including alerting those in the immediate area, as well as notifying the central communication center of the installation.

2. A well-trained local emergency organization is developed to evacuate those in the immediate area to a point of safety, to detain personnel at that point of safety until a radiation survey is made, to notify the communication center and, if possible, make recommendations as to the immediate action that should be initiated, to account for all personnel, and to evaluate the situation and take appropriate action.

In addition to such local or line organization emergency groups, emergency service and assistance groups should be available. For the most part, these groups would be made up of specialists, including medical men, health physicists, analytical chemists, firemen, and utility men, but plans must also be made for obtaining emergency supplies, transportation, and labor.



DUTIES AT EACH LOCATION:

I. AT SCENE OF EMERGENCY
LINE ORGANIZATION
1. EVACUATE ALL PERSONNEL IN IMMEDIATE AREA.
2. NOTIFY COMMUNICATION CENTER.
3. MAKE NECESSARY OPERATIONAL CHANGES AND AID OTHERS IN AVOIDING THE RADIATION.

II. AT COMMUNICATION CENTER
EMERGENCY COORDINATOR
1. EVACUATE PERSONNEL FROM OTHER POTENTIALLY HAZARDOUS AREAS.
2. EVALUATE SITUATION.
3. COORDINATE EMERGENCY.
OTHER EMERGENCY SERVICE PERSONNEL
1. OPERATE COMMUNICATION CENTER.

III. AT ASSEMBLY POINT
HEALTH PHYSICS PERSONNEL
1. MONITOR ALL EMPLOYEES.
2. DISPATCH PATIENTS TO CLEANING STATION, DISPENSARY, ETC., ACCORDING TO THEIR NEEDS.
3. ESTABLISH PERSONNEL TRAFFIC PATTERN WITHIN ASSEMBLY POINT TO PREVENT UNDUE SPREAD OF CONTAMINATION.
LINE ORGANIZATION
1. ACCOUNT FOR ALL EMPLOYEES AND CONFINE THEM UNTIL RELEASED BY HEALTH PHYSICS.
2. RENDER NECESSARY FIRST AID TO INJURED.
3. EVALUATE SITUATION AS TO POTENTIAL DANGER FROM THE UNATTENDED PROCESS AND EQUIPMENT. PLAN COURSE OF ACTION.
4. HAVE ALL RELEASED EMPLOYEES FILL OUT A HEALTH PHYSICS REPORT.
OTHER EMERGENCY SERVICE PERSONNEL
1. FURNISH RADIO COMMUNICATION.
2. TRANSPORT PATIENTS AS DIRECTED BY HEALTH PHYSICS.
3. DISTRIBUTE HEALTH PHYSICS QUESTIONNAIRES TO ALL EMPLOYEES WHEN RELEASED.

IV. AT CLEARING STATION
MEDICAL PERSONNEL
1. OPERATE STATION.
2. SUMMON ADDITIONAL MEDICAL PERSONNEL AS NEEDED.
HEALTH PHYSICS PERSONNEL
1. ASSIST MEDICAL PERSONNEL BY MONITORING PATIENTS REACHING THIS STATION.
2. CONSTANTLY MONITOR STATION FOR UNDUE SPREAD OF CONTAMINATION.
3. COLLECT PATIENTS' FILM BADGES AND ARRANGE FOR PROCESSING.
OTHER EMERGENCY SERVICE PERSONNEL
1. FURNISH RADIO COMMUNICATION.
2. FURNISH TRANSPORTATION, INCLUDING AMBULANCE SERVICE.

V. AT DECONTAMINATION STATION
HEALTH PHYSICS PERSONNEL
1. MONITOR ALL PATIENTS AND CONTROL CONTAMINATION IN THIS FACILITY.
2. COLLECT FILM BADGES.
3. HAVE PERSONNEL FILL OUT HEALTH PHYSICS QUESTIONNAIRE WHEN RELEASED FROM STATION.
OTHER EMERGENCY SERVICE PERSONNEL
1. OPERATION STATION TO INCLUDE
a. FURNISH CLEAN CLOTHING AS NEEDED.
b. WRAPPING AND MARKERS FOR DISCARDED CLOTHING.
c. FURNISHING BATHHOUSE AND OTHER TYPES OF SUPPLIES.

Figure 2—Operational procedure for a radiation incident (immediate action plan).

In general, the emergency service and assistance groups should be assigned specific responsibilities and should have made and practiced their detailed procedures for supporting the over-all installation plans when called upon. Finally, an installation-wide emergency director is required to ensure that all emergency groups involved function as a team.

A flow sheet of personnel and information for handling the initial phases of a radiation incident is presented in Fig. 2. This plan is applicable to installations of any size by modi-

fication or expansion. The location of assembly points, decontamination centers, and medical stations should be as clearly and definitely specified as are the individual duty assignments. (W. Y. Gissel)

Radiation Incident Reports

Facility licenses carry a specific provision that the AEC be immediately informed, in writing, of any incident and any possibly unsafe condition relative to the operation of the facility. In addition, licensees are required to make an annual report of operating experience and to keep records on the release or discharge of any radioactivity beyond the effective control of the licensee. These provisions are not completely definitive, however, and there is always a question of judgment involved as to what constitutes an incident and what indicates a possibly unsafe condition.

This subject was discussed by C. K. Beck, Chief of the Hazards Evaluation Branch, during the hearing to consider amendments to the General Electric Company operating license for the Vallecitos Boiling-Water Reactor (VBWR).[28] In the course of the hearing it was brought out that various incidents took place during 1957, 1958, and 1959 that were not reported to the AEC because, in the Company's opinion, they did not amount to indications of an unsafe condition. However, following a July 1959 request from the AEC for information on one incident, the Company agreed to report all incidents. Beck stated that reportable incidents (1) need not include all activity releases, (2) might be difficult to define quantitatively, and (3) could be qualitatively described in candid discussions between the AEC and the applicant (or licensee).

In order to improve the reporting and investigation of radiation incidents, the AEC has established definite procedures and responsibilities, which are incorporated in Chapter 0703 of the AEC Manual.[29] The new procedures, which were approved Mar. 16, 1960, and were published in April, establish a single basic system applicable to investigations of all types of incidents and set forth policies, standards, and responsibilities for reporting and investigating incidents in AEC contractor and licensee operations. Under the new procedures the Division of Compliance will receive concurrent notification of incidents reported immediately to AEC headquarters, as well as reports on management followup on findings and recommendations. The categories of incidents that require immediate notification to AEC headquarters are listed.

Provisions of the new procedures which are of particular interest in the prevention of recurrence of an incident are quoted below:

0703-022 Investigation of incidents shall be initiated by responsible AEC officials and shall provide information to:

. . . . . .

c. determine corrective action appropriate to minimize or preclude similar incidents;

. . . . . .

e. assist in the improvement of AEC policies, standards, and regulations, and of operations thereunder;
f. prepare a written report.

0703-041 Managers of Operations are responsible for:

. . . . . .

c. reporting fully to the responsible operating division and to the Division of Inspection the management actions taken or to be taken on findings and recommendations contained in investigative reports to Headquarters.

0703-043 The Division of Inspection is responsible for:

. . . . . .

b. the conduct of all Headquarters investigations except those assigned to the Division of Compliance;
c. conducting independent investigations under certain conditions.

0703-045 The Division of Compliance is responsible for:

. . . . . .

b. informing the Assistant General Manager for Regulation and Safety of the occurrence, investigation, and management actions taken regarding incidents.

Among the standards for investigation of incidents by managers of operations are:

5. As a minimum, investigative reports should include:

. . . . . .

b. an orderly presentation of the facts relative to the incident;
c. a clear statement of the conclusions reached;
d. clear and specific recommendations which will make effective disposition of all phases of the case;

6. ... A record shall be maintained of management actions taken on investigation findings and recommendations.

(L. A. Mann)

Evaluation of Reactor Hazards

A qualitative method for the estimation of the magnitude of the hazard associated with the operation of a nuclear reactor has been proposed by Siddall in a recent paper.[30] In terms of money and lives per unit time, Siddall has equated the upper limit of the permissible hazard that arises because of a particular operation to the expected gain in gross national product and in over-all increased life expectancy. Implicit in the analysis is the definition of the term "hazard." It is the intent of the following discussion to make this definition explicit and to suggest further criteria that may be used for the purpose of arriving at a numerical value of the magnitude of a hazard. The reader is warned not to expect to find here a solution to the problem of assigning numerical values to hazards. Rather, it is hoped, by characterizing the problem in somewhat more exact terms than are usually encountered, to suggest possible methods of monetary evaluation.

Definition of Hazard

The word "hazard" may be loosely defined to mean "exposure to risk of loss or injury." Everyone is constantly exposed to various kinds of risks, and what is meant when it is said that some operation or course of action is hazardous is that the probability of injury resulting therefrom is large or that the injury, should it occur, will be serious. In order to give quantitative meaning to the word "hazard," consider a course of action which, if pursued, has a finite probability of one and only one type of injury or loss. If this course of action results in injury or loss, that event will be called an "accident." If the total cost (dollars, for example) that would accrue should the accident occur is denoted by D, the parameter D will be called the "dollar damage." Let r denote the "risk," i.e., the probability per unit time (years, for example) that the course of action will result in an accident. Then, in this simple case, the "dollar hazard" in dollars per year, denoted by H_d, may be defined as

$$H_d \equiv Dr$$

In some accidents loss of life may occur, and presumably a dollar value could be placed on human life and included in the value of D. For reasons that will become apparent, however, it seems better to include only the cost of property loss, loss of working time, insurance payments, medical bills, etc., in the value of D. In order to account for loss of life, a second type of hazard, the "mortality hazard," denoted by H_m, may be defined. If the expected loss of life from the accident is M lives, then the mortality hazard in lives per year may be defined as

$$H_m \equiv Mr$$

With the use of these relations, it is possible, in principle, to characterize the hazard which arises from the given operation in terms of the two numbers H_d and H_m.

Obviously the situation described above is quite artificial, since it was assumed that only one type of accident could result from a given operation or course of action. There are, of course, many different types of accidents which can be associated with a given operation, and moreover, there are many different ways of classifying them. Thus one is led naturally to the notion of "partial hazards." If, for example, the course of action under consideration were the operation of a motor vehicle, several different types of accident could be postulated, and a partial hazard could be defined for each type. The total hazard associated with the operation of the motor vehicle would be the sum of the partial hazards. The H_d and H_m relations would then become

$$H_d = \sum_{j=1}^{n} H_{d_j} = \sum_{j=1}^{n} D_j r_j$$

and

$$H_m = \sum_{j=1}^{n} H_{m_j} = \sum_{j=1}^{n} M_j r_j$$

where j refers to the jth type of accident and ranges over all possible types of accidents as-

sociated with the operation in question. In a typical case, j might range from 1 to 7. The seven types of accidents and H_m values for each, based on data available from the National Safety Council,[31] are listed in the following tabulation.

Type of accident	H_m, lives per year ($\times 10^4$)
Striking a pedestrian	1.142
Collision between vehicles	1.845
Overturning or running off road	1.918
Collision with railroad train	0.177
Collision with bicycle	0.070
Collision with fixed object	0.249
Other	0.016
Total	5.417

Hence the total mortality hazard which arises from the operation of a single motor vehicle is, on the average, 5.417×10^{-4} lives per year. The corresponding H_d value, or dollar hazard, is $81.78 per year.

In this illustration the partial hazards were obtained a posteriori from accident statistics and represent an average value for all operating motor vehicles in the United States during the year 1958. The task of obtaining these figures a priori for a single vehicle would, if even possible, be extremely complex. In the case of the operation of a nuclear reactor, the statistics that would be needed to obtain reliable data on hazards are not available, and thus some sort of a priori calculations must be made. Actually the task of computing reasonable values of M_j and the D_j is not too formidable provided the possible accident situations can be listed. The difficulty lies in finding the values of the r_j and, to a lesser extent, in listing all the possible types of accidents. The latter is obviously closely related to the former but is probably more susceptible to intelligent rationalization. The customary way of side-stepping this problem has been to examine in detail those accidents for which the values of D_j and M_j are highest and for which, by some subjective process of reasoning, it has been determined that r_j is significant. This is the so-called "maximum credible accident" approach. It has the weakness, also pointed out by Siddall, that, although it selects for study the most spectacular accident, it neglects the fact that an accident which may cause less damage, but which is more likely to occur, can very easily constitute a greater hazard than the maximum credible accident. Moreover, once this choice of the "worst case" is made, it may obscure other more subtle possibilities.

Acceptable Hazards

Every operation or course of action has associated with it some finite hazard, and, in principle, if the upper limit of the hazard, which can be chosen for each operation, is exceeded, that operation would be rendered undesirable. The upper limit would, of course, vary with the individual making the choice. For this discussion the "acceptable hazard" associated with any given operation is considered to be the hazard that is of a magnitude which is not more than the public is willing to condone. Before turning to a discussion of the possibility of finding ways to estimate r_j some consideration must be given to the question of what the values of the acceptable hazards associated with the nuclear energy industry are likely to be.

It is probably not too much of a generalization to assert that the magnitude of the hazard associated with a given operation that is acceptable to the public will be proportional to the amount of benefit or pleasure which the public believes it will realize from the operation. Since this belief depends upon the state of public knowledge with respect to the operation, the value of the acceptable hazard may change with time. The problem is further complicated by the fact that bizarre or spectacular accidents appear to be more feared than do commonplace accidents. It is an odd characteristic of the human race that its members prefer to be maimed or killed by familiar and well-proven methods. Infrequent accidents in which loss of life is high appear to be less tolerable than more frequent accidents in which loss of life is low, even though the mortality hazard may be lower in the former than in the latter case. Conversely, infrequent accidents accompanied by high dollar loss seem to cause less indignation than do more frequent low-dollar-loss accidents. Whether or not this behavior of the public is illogical is not germane to the issue at hand. The fact is that it exists, and it must be considered in determining the magnitudes of the acceptable hazards. It might be remarked, parenthetically, that an investigation of the re-

action of the public to situations which are at the same time hazardous and beneficial might well produce some interesting and useful results.

There is one further distinction which must be made at this point; that is, the distinction between hazards which affect only persons and property directly associated with the operation from which the hazards arise and those which affect persons and property not associated with that operation—the innocent bystander. For example, the hazards inherent in the operation of a coal mine affect primarily only those persons employed in this operation and the property of the mining company; on the other hand, the hazards which arise in the manufacture and storage of anhydrous ammonia may affect not only the employees and the plant property but also all persons and property in the immediate neighborhood of the plant. Those hazards which affect only the employees and the plant property will, as is usual, be called "occupational hazards" and will be denoted by the superscript o. The hazards which affect off-plant property and personnel will be called "public hazards" and will be denoted by the superscript p. Thus the dollar hazard may be written as

$$H_d^o = \sum_{j=1}^{n} D_j^o r_j$$

$$H_d^p = \sum_{j=1}^{n} D_j^p r_j$$

$$H_d = H_d^o + H_d^p = \sum_{j=1}^{n} (D_j^o + D_j^p) r_j$$

Similar expressions can be written for the mortality hazards.

Only the public hazards will be considered in the following, not because it is felt that the occupational hazards are unimportant, but rather because less is known about the hazards to the general public. In considering the public hazards, such long-range hazards as the increasing concentration of fission products and other toxic substances in the various environmental media will be omitted, even though such conditions may well turn out to be among the most dangerous associated with development of the nuclear industry, and attention will be confined to injuries and losses which could occur in a relatively short period of time.

The question of the magnitude of the acceptable external hazard arising from the operation of a nuclear reactor can, in the last analysis, be answered by the public only. It seems logical to suppose that the greater the benefit to be expected from nuclear energy, the greater will be the acceptable risk. As was suggested above, however, the public is not constrained to act in a logical fashion. Moreover, no omniscient agency is available to define what a logical fashion consists of. For the present it appears that it will not be acceptable in the nuclear industry to greatly exceed the magnitude of the public hazards generally found in well-established industries. One possible criterion is to require that the public hazard from a reactor shall not exceed the average for all other types of industrial operations.

Another method would be to relate the public hazard to some familiar operation. In doing this, care must be taken because the analogies are often not complete. At the risk of falling into this very error, consider, for example, the requirement that the public hazard introduced by the construction and operation of a reactor shall not exceed the total hazard arising from an increase of 1 per cent in the number of motor vehicles operating in the vicinity of the reactor location. Then, if there were 300,000 automobiles in the area considered, it would be required that H_m^p be less than 1.625 lives per year and H_d^p be less than $\$245,000$ per year. The prospect of this annual loss to the community would certainly not prevent 3000 persons from buying automobiles. But would it cause serious opposition to the construction of a reactor?

If we take the simple case in which there is only one type of external hazard, and we assume that the corresponding accident would cause the loss of 100 lives and $10 million in property damage, then, on the foregoing basis, we must have $r < 0.01625$ to satisfy the condition on H_m^p and $r < 0.0245$ to satisfy the condition on H_d^m. By taking the smaller of these, $r < 0.01625$ per year, we see that both conditions are satisfied, and the expected accident frequency must be less than 1 per 62 years.

Hazards Probabilities

So far we have carefully avoided a discussion of methods for a priori evaluation of r_j. It seems reasonable to conclude that, given the description of an accident, the results can be estimated with some degree of confidence. On the other hand, the tabulation of the spectrum of

possible accidents, together with their probability of occurrence, is of the utmost complexity, if it is, indeed, attainable at all. It is the primary purpose of these remarks to focus the attention of the reader on the fact that the determination of these probabilities is the central problem of hazards analyses. It is perfectly clear that, if either of the factors D or r is zero, then no hazard exists. It is equally clear that, in the nuclear industry, the values of D will often be very large. Therefore, in order to reduce the hazard to an acceptable value, it will be necessary to make r very small. It is thus important to develop ways of evaluating r.

Some approaches to this problem have been made, notably by Kahn[32,33] and by Freedman and Langevin.[34] Kahn has examined[32] the situation in which safety instrumentation is inspected periodically and has studied the probabilities of failure of the instrumentation, followed by an operating system failure, within the inspection period. In the second paper[33] Kahn discusses the optimization of the design and maintenance of such a system from an economic standpoint. Freedman and Langevin[34] have constructed a simplified stochastic model of reactor operations that focuses attention primarily on the interaction of human error with the components of the system. Although these papers, to the best of the reviewer's knowledge, are the first to give recognition to the problem and the first to attempt to attack it from a probabilistic standpoint, it is clear that a great deal of work remains to be done.

One approach which may prove fruitful is to abandon any attempt to assign absolute values to r and, having in some subjective manner picked a standard, to attempt to evaluate r relative to this standard. For instance, the risk of a startup accident will, all other things being equal, be reduced by a factor of 2 if the number of startups is halved. The risk of a reactor runaway is changed in a predictable manner by a change in the amount of available excess reactivity. If the damage factor in a partial hazard under consideration is proportional to the power level, an increase by a factor of 1.5 in power would require a decrease by the same amount in the corresponding risk. All other things being equal, this might take the form of a three-channel safety circuit rather than a two-channel circuit. It should also be noted that increases in D_j for one reason can be offset by decreases for some other reason, thus eliminating the need for a change in the corresponding r_j.

Similar comparisons can be made with respect to other hazards; however, in the case of mortality hazards, it may be necessary to make further concessions to the psychological factors affecting public acceptance of these hazards. It is true that 1 times 100 is equal to 100 times 1; however, it may be that one accident per 100 years which causes 100 deaths is considered less acceptable to the public than one death per year for 100 years. If this turns out to be the case, it may be necessary to reduce the allowable partial hazard corresponding to large values of M_1 as the value of M_1 increases.

Conclusions

The advantages of a scheme such as that proposed above are numerous. The introduction of quantitative definitions of the various hazards provides a nomenclature that permits discussion of the subject within a common frame of reference. The use of quantitative ideas, even though the parameters involved cannot be fully evaluated, permits the separation of the causes of hazards into component parts that are susceptible to study. The use of some procedure for estimating the values of r_j provides the designer with at least a semiquantitative way of appraising the relative magnitude of the hazards associated with his design and should facilitate his discovery of the parameters most likely to affect the hazards. Finally, the proposed scheme permits an examination of the causes and results of accidents that can be kept free from emotional influence. The main disadvantage, of course, is a lack of understanding of ways in which to estimate the risks. In addition, there is the ever present danger that the designer will place too much trust in his calculations.

Most of the work on hazards which has been done to date has been directed toward evaluating the results of various accidents, i.e., assessment of D_j and M_j. With the exceptions of the papers cited, the reviewer is not aware of any serious attempt to study quantitatively the probability that dangerous situations will indeed arise. If a reasonably accurate method of obtaining values of the risks can be devised, it should then be possible to elevate the business of evaluating reactor hazards to the rank of a science. (F. T. Binford)

Basic Considerations of Radiation Protection

The bases for determining standards of radiation protection and the designation of appropriate agencies for establishing and enforcing such standards are of vital concern to nuclear industry and research. The problems of radiation protection have been with the professional users of radiation for over half a century but are now engaging the attention of the public and of the government to an increasing degree. The Congressional hearings of the Joint Committee on Atomic Energy from May 24 to June 3, 1960, were largely concerned with these problems and will doubtless be educational for the public, as well as a major influence in shaping the policy of the government.

During the first quarter of this century, radiation protection was largely the responsibility of the user, and except for scientific laboratories, the users were the small but increasing number of radiologists who used X rays for diagnosis and therapy. In 1928 the forerunner of the present International Commission on Radiological Protection (ICRP) was formed, and various national committees, such as, in the United States, the National Committee on Radiation Protection (NCRP), came into existence at about the same time. These were professional groups which, without legal authority, issued recommendations concerning radiation protection. By 1935, when experience with radium had emphasized the subtle nature of the hazard and demonstrated the existence of long-delayed effects of radiation, these professional groups were already well enough established to meet the challenge adequately. The limit on body burden published in 1941 by these professional groups is still the standard for occupational exposure to radium.

Each of the four articles reviewed here[35–38] is concerned with one or another aspect of the question of the proper bases for the determination of radiation-protection standards. The report of the *Ad Hoc* Committee of NCRP[35] on "Somatic Radiation Dose for the General Population" is the document of main interest. The other three articles[36–38] are briefly reviewed to illustrate the current public interest in basic policies of radiation protection and the treatment afforded these issues by the press.

The development of atomic energy vastly increased the complexity of the problems. The number and variety of sources, the intensity of the radiation, and the number of people potentially exposed to radiation all increased many fold. Even during the period of secrecy imposed by the war, it was largely the experience and judgment of these professional groups that defined standards applicable to the new sources. After the war, when atomic energy emerged into public view and began to be applied for scientific, industrial, and medical uses, the ICRP and NCRP elaborated their recommendations to keep pace with the needs. It is a tribute to these groups that, throughout almost 15 years of rapid development and expansion of this technology, their recommendations have been the basis for radiation protection, and during this time the atomic energy industry has been one of the safest of modern industries.

By 1957 the number of people potentially exposed reached a maximum when it was realized that radioactive fallout was distributed over the entire globe. Radiation protection then became a problem of general interest to the public, and, when it became apparent that estimates of the hazard lacked the precision and finality the public generally associates with scientific pronouncements, the interest became more than a little tinged with concern. During the last three years, a succession of Congressional hearings, the space devoted to radiation hazards by the press, and the organization of groups to protest or urge a point of view have indicated a continual growth of awareness of the problem in governmental circles and also on the part of the public.

The 1958 recommendations of the ICRP[39] gave rather detailed suggestions concerning occupational exposure, exposure of special groups, and also exposure of the populace. The ICRP suggested that the genetic dose from all sources other than medical practice or natural background radiation be limited to 5 rem to age 30 years. This limit was based on estimates of the changes in the mutation rate such a dose might produce and on the consequences foreseen by the geneticists. The ICRP did not suggest a dose limit based on somatic damage, but they did suggest limits on environmental contamination which would control this damage. In the case of somatic damage, the hazard is largely to the individual rather than to the race, as in the case of genetic damage, and little

information is available concerning somatic effects in man from doses of radiation of the order of background radiation. The 1958 suggestions were somewhat modified[40] by the ICRP in July 1959, but the modifications do not indicate the scientific or other considerations that were used to determine the dose limits based on somatic damage. Meanwhile, the NCRP adopted[41] essentially the same limits on genetic dose as those recommended or suggested by the ICRP but appointed an *Ad Hoc* Committee to suggest proper bases and review the question of dose limits based on somatic effects.

The report of the NCRP *Ad Hoc* Committee[35] states that there is no firm dose-effect relation known in the range of dose that is of interest, i.e., there is almost no quantitative and little qualitative evidence of the biological effects that can be attributed to doses of radiation of the order of background radiation. The Committee considers that it is conservative to assume a linear relation and then extrapolate from the effects of high doses to the effects of lower doses. On this basis it follows that there is no threshold below which damage is nonexistent, and thus all radiation exposure must be considered as having some harmful effects.

It follows that no exposure is justified unless there is the expectation of compensating benefits, and the greater the benefits, the greater the risk society or the individual may be willing to incur in order to have these benefits. This is a truism in the case of medical exposure, where very large doses are frequently accepted in order to treat a malignancy which otherwise might be fatal. Whether the hazard of the treatment is overbalanced by the benefits is a matter for the individual and his doctor to decide, and the public is not involved in a major way. The patient and his doctor must "weigh the expected benefits against the hazard of the radiation" and decide accordingly.

Problems of waste disposal and environmental contamination often present situations in which the benefits and the hazards are not confined to the same individuals or groups of individuals. Thus, although atomic power may well prove to be a major step toward a fuller and richer life for modern man, the hazard from the waste products may be largely confined to certain groups which happen to live near a given site. Similarly, weapons testing may be a necessary condition for the preservation of the American way of life, but the fallout material is not confined to the United States and it is not distributed uniformly. Thus the hazards may not be commensurate with the benefits in the case of certain individuals or groups. Actually this situation is present to some degree in most areas where hazards must be regulated, and it is clear that the government is the agency for finally deciding the questions thus raised.

It follows from these premises that limits may be different for different applications, i.e., greater benefits may justify a greater level of risk. Since it is difficult to assess either the benefits or the risks in any very precise way, it is necessary to have a basis for exposure such that elaborate studies and evaluations are not necessary. The *Ad Hoc* Committee considers that the level of background radiation supplies such a basis. Man has evolved in the presence of this degree of exposure, and therefore its consequences do not seem to be catastrophic. Moreover, since it varies by factors of 2 or 3 over wide areas and since any consequences of this higher degree of exposure have thus far escaped detection, it is indicated that a doubling of the background dose, or perhaps even a tripling, may be expected to have no immediately discernible effects. Thus the *Ad Hoc* Committee recommends that "the maximum permissible dose of man-made radiation (excluding medical and dental sources) should not be substantially higher than the background level of natural radiation without a careful examination of the reasons for higher values."

The report also points out that by the same linear hypothesis a situation does not change from "safe" to "dangerous" when any selected dose level is exceeded. In the absence of a threshold, there is a gradual increase of risk as the dose increases. Thus the words "substantially higher than background" do not imply a sharp criterion. As remarked above, the background varies by factors of 2 or more with location, and this also implies a fairly wide band of values rather than a single number.

The *Ad Hoc* Committee was not charged with the task of selecting a dose limit. Thus the report only indicates an approach to the problem. Although this approach cannot be said to be new, it has seldom been presented with so much emphasis. The NCRP has accepted the report and has published it, but, since it presents only a philosophy of protection rather than specific guides, it has not been adopted as an official recommendation of the NCRP. The

report has been turned over to Subcommittee I of NCRP, which is charged with the selection of basic limits of exposure.

The report of the *Ad Hoc* Committee will serve a useful purpose in that it exemplifies one particular approach to the problem, and, in the absence of data on biological effects, it is perhaps the only quantitative basis now available. It has several severe limitations so far as practical use is concerned. Although the report has recognized that higher dose levels may be appropriate and are to be set by "balancing the benefits and the risks," it does not indicate how either the benefits or the risks are to be assessed. Whether the background radiation is to provide any guidance for setting these higher levels is not explicitly set forth. On the linear hypothesis the risk would increase proportionally with dose, but, since the initial dose must be attributed to background radiation, the linear hypothesis does not provide a quantitative guide for estimating such risks. It is generally considered that only a fairly small fraction of "naturally occurring" malignancies can be attributed to background radiation.

The report does not distinguish between exposure of single organs or exposure of major portions of the body, although experiments have indicated rather marked differences in the effects to be expected. Likewise, the report is rather vague with respect to whether the limit applies to each individual or is to be interpreted as an average over a population or group. Undoubtedly, when the NCRP seeks to apply the principles enunciated here in setting levels of dose for various uses and groups of people, these questions must be resolved.

The *Ad Hoc* Committee report does not consider who is to undertake the task of balancing the benefits and risks. There has, however, been considerable discussion of this point by other individuals and groups. It has been argued that professional groups, such as NCRP and ICRP, were the appropriate agencies when uses of radiation were largely in the hands of a tightly knit professional group, such as the radiologists. Since early scientific and industrial uses of radiation did not result in exposures of large numbers of people, the central problem was that of a profession. Over the past 15 years, however, radiation protection has been effectively the concern of the AEC, although it could only be responsible for the actions of itself, its contractors, and its licensees. The AEC took the recommendations of NCRP and adapted them to its specific problems, but many industrial, scientific, and medical uses of radiation were not within its jurisdiction. The fact that the AEC is responsible both for promoting the use of radiation and for regulating its use to ensure a proper degree of protection has been a source of criticism. Currently the AEC is offering the state governments a chance to assume some responsibility for radiation protection. Recent articles[36,37] have indicated public questioning of these and related matters, and a summary of various points of view concerning the proper governmental agency to be primarily responsible for radiation protection has been issued by the Atomic Industrial Forum.[38]

In October 1959 the Congress established the Federal Radiation Council,[42] and charged this cabinet-level body with "coordinating the policy of the various governmental agencies in matters of radiation protection." Although it is too early to foresee just what role each will play, it seems clear that the federal government, the state governments, and even municipal governments will eventually share responsibility for radiation protection. On the federal level, it seems likely that the Public Health Service, the Department of Defense, and others may eventually share responsibility along with the AEC.

The background material and points of view which eventually led to the formation of the Federal Radiation Council were discussed in a *Science* article.[36] The question of the proper agency to determine limits for population exposure and to monitor for their enforcement of the limits was considered, and the AEC's Project Sunshine, which is concerned with the fallout problem, was described. The record of the Public Health Service in the field of radiation protection was cited. Curiously, the *Science* article did not mention the cooperation of the Bureau of Standards and the NCRP that has been perhaps the most influential effort in this field (see reference 41).

Another *Science* article,[37] entitled "Reports Disagree on Radiation Hazards," directs attention to certain conflicts which are supposed to exist between recommendations of the NCRP and the ICRP without doing anything to clarify or even identify the issues. The specific reports and recommended limits in question are

only loosely identified. The writer seems unaware that the recommendations of the ICRP and NCRP so far as occupational exposure is concerned are basically similar. As to limits for exposure of populations, the recommendations of NCRP and ICRP on genetic dose are virtually identical. For somatic dose, the NCRP, as mentioned above, appointed its own *Ad Hoc* Committee to consider the question, but the Committee had not submitted its report at the time the *Science* article was published. Therefore the reported "conflict" on matters of somatic population exposure was in anticipation of the NCRP report. Thus far the conflicting recommendations of the NCRP and ICRP have been largely a matter of the different dates of publication of the reports of these groups. (W. S. Snyder)

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Section II ACCIDENT ANALYSIS

Autocorrelation Functions and Operational-Safety Analyses

The use of autocorrelation functions in noise analysis has been conventional in electrical engineering and related fields for many years; the first symposium[1] held in this country on applications of autocorrelation analysis to physical problems was held in June 1949. Application to the analysis of actual reactors has been relatively recent, however, and it was apparently accelerated by the work of Moore.[2]

The autocorrelation function can be used in analyzing reactor "noise," and it permits a determination of the relative magnitude of the transfer function as the transfer-function parameter travels the imaginary axis. Where applicable, this method of analysis permits the evaluation of the effective delayed-neutron fraction divided by the mean lifetime of prompt neutrons. For unknown transfer functions, it determines their shape; in particular, it detects sharp resonances in the transfer function. Alternatively, if the mathematical model is known, evaluation of coefficients in the equations can be performed with knowledge of the transfer-function amplitude as the function parameter takes on different values; or, if all coefficients are known, information can be obtained about the input noise or reactivity disturbance. To illustrate its application, noise analysis has been used to determine specific reactor parameters for several reactors at the Argonne National Laboratory (ANL),[3] for the Sodium Reactor Experiment (SRE) and the Kinetic Experiment on Water Boilers (KEWB),[4] and for the Experimental Boiling-Water Reactor (EBWR).[5]

Operational-Safety Application

The application of noise analysis to operational safety is associated with the transfer function as a function of operating conditions. As changes occur in reactor operating conditions, the transfer function of the system may also change, and this change may be indicative of certain undesirable conditions being present within the reactor. For example, in the case of a homogeneous fluid-fuel reactor, deposition of fuel on surfaces within the core is undesirable; however, such deposition is difficult to measure physically, and a relatively long time [about a day on a routine basis for the Homogeneous Reactor Test (HRT)] is required to obtain an indication of deposition from the results of chemical analyses. By noise analysis, it may be possible to obtain quickly some of the characteristics of the system transfer function at various operating times, and thereby determine whether undesirable physical changes are occurring within the reactor. Some preliminary analyses of this nature have been performed by Hirota,[6] and they will be discussed later. The physical changes associated with changes in transfer-function characteristics may have to be determined empirically, but, once established, specific characteristics can be associated with desirable or undesirable operating conditions. Even if correspondence with specific physical changes were not established, a marked change in transfer-function characteristics could indicate an abnormal condition that should be investigated with care.

Background

In order to appreciate what information reactor noise analysis can provide, it is pertinent to review reactor stability, transfer functions, characteristic equations, Nyquist's stability criteria, and frequency-response plots.

Stability. In specifying the nuclear design of a reactor, it is desirable that the criteria be such that the reactor will be stable under oper-

ating conditions. As applied to reactors, stability refers, in general, to those characteristics that will cause the reactor power to approach an equilibrium value following a system disturbance. In this discussion, a stable system is one in which there is no buildup in power oscillations following a reactivity addition.

In studying reactor stability, the appropriate time-dependent equations of motion are examined. Although equations that describe reactor systems are basically nonlinear, stability is usually determined on the basis of linearized equations of motion. There are two fundamental reasons for this approach: (1) the behavior of the nonlinear system approaches that of the linear one if the system disturbances are sufficiently small, and (2) linear equations can be analyzed in a more generalized manner than can nonlinear equations, and thus more information concerning system behavior can be obtained. In the following discussion, it is assumed that, for the purpose of stability analysis, the reactor system is approximated by a set of linear equations with constant coefficients.

Characteristic Equations and Transfer Functions. The reactor parameters, or reactor characteristics, that are involved in the determination of system stability appear in the so-called "characteristic equation." In order for a system to be stable, it is necessary that the characteristic equation have no root with a positive, real part.

For purposes of illustration, the conventional linearized kinetic equations will be used and written as

$$\frac{dx}{dt} = \left[\frac{\delta k_e(t) - \beta x}{l}\right] + \sum_i \lambda_i c_i \qquad (1)$$

and

$$\frac{dc_i}{dt} + \lambda_i c_i = \frac{\beta_i}{l} x \qquad (2)$$

where c_i = latent power associated with delayed-neutron precursors of ith group, normalized to power at time zero
$\delta k_e(t)$ = reactivity
l = prompt-neutron lifetime
t = time
x = reactor power normalized to its value at time zero
β = fraction of fission neutrons which are delayed
β_i = fraction of fission neutrons in ith group of delayed neutrons, $\sum_i \beta_i = \beta$
λ_i = decay constant associated with ith group of delayed-neutron precursors

Stability is determined by the time behavior of x following a reactivity disturbance. Since deviations from initial values are of primary interest, Eqs. 1 and 2 shall first be modified by letting

$$\zeta \equiv x - 1 \qquad (3)$$

and

$$v_i \equiv c_i - c_i(0) = c_i - \frac{\beta_i}{l\lambda_i} \qquad (4)$$

Equations 1 and 2 then become

$$\frac{d\zeta}{dt} = \frac{\delta k_e(t)}{l} + \sum_i \lambda_i v_i - \frac{\beta}{l}\zeta \qquad (5)$$

and

$$\frac{dv_i}{dt} + \lambda_i v_i = \frac{\beta_i}{l}\zeta \qquad (6)$$

In order to obtain $x(t) = \zeta(t) + 1$, it is convenient to solve Eqs. 5 and 6 by means of the Laplace transform. Letting

$$\overline{\zeta} \equiv \mathcal{L}[\zeta(t)] \equiv \int_0^\infty e^{-st}\zeta(t)\,dt$$
$$\overline{v}_i \equiv \mathcal{L}[v_i(t)]$$
$$\overline{\delta k_e} \equiv \mathcal{L}[\delta k_e(t)]$$

the transformed equations give $\overline{\zeta}$ as

$$\overline{\zeta} = \frac{\dfrac{\overline{\delta k_e}}{l}}{s + \dfrac{\beta}{l} - \sum_i \dfrac{\lambda_i\beta_i}{l(s+\lambda_i)}} \qquad (7)$$

The value for $\zeta(t)$ is obtained by taking the inverse Laplace transform of $\overline{\zeta}(s)$, namely,

$$\zeta(t) = \mathcal{L}^{-1}[\overline{\zeta}(s)] = \frac{1}{2\pi i}\int_{b-i\infty}^{b+i\infty} e^{st}\zeta(s)\,ds \qquad (8)$$

where b is chosen such that all singularities of

$\zeta(s)$ lie to the left of the line having b as its real part. The singularities of interest will occur at the zeros of the denominator term of $\zeta(s)$ [the singularities associated with $\overline{\delta k_e}(s)$ will be determined by the reactivity driving function, and, as such, they do not influence stability]. Therefore, in evaluating the integral term in Eq. 8, the values of s associated with the resulting e^{st} terms will be those which satisfy the following equation:

$$s + \frac{\beta}{l} - \frac{\lambda_i \beta_i}{l(s + \lambda_i)} = 0 \qquad (9)$$

Hence, $\zeta(t)$ can be written in the form

$$\zeta(t) = \sum_j A_j \, e^{s_j t} \qquad (10)$$

where s_j is the jth root of Eq. 9 and A_j is the constant associated with the $e^{s_j t}$ term. The behavior of $\zeta(t)$ is thus determined by the values for s_j. If none of the s_j values have roots with a real, positive part, then $\zeta(t)$ cannot build up with time; whereas, if any of the roots has a positive, real part, $\zeta(t)$ will increase indefinitely with time following a disturbance. Equation 9 is called the characteristic equation for the system considered, and, as indicated above, the time behavior of $\zeta(t)$ is determined by the character of the roots.

Equation 7 can be written also as

$$W(s) = \frac{\overline{\zeta}(s)}{\overline{\delta k_e}(s)} = \frac{1}{ls + \beta - \sum_i \frac{\beta_i \lambda_i}{(s + \lambda_i)}} \qquad (11)$$

Written in this manner, it represents the Laplace transform of the output, $\overline{\zeta(s)}$, divided by the transform of the input, $\overline{\delta k_e}(s)$. This ratio is defined as the "transfer function" and is designated here as $W(s)$. It also is the inverse of the characteristic equation of the system. Hence the transfer function is directly related to the stability of a system.

Nyquist's Stability Criteria. After the transfer function, or characteristic equation, has been obtained, the next step is to determine the character of the associated roots as a function of the coefficients which appear in the equation. All possible roots having real, positive parts need to be considered; this is done by studying the right-half plane (shaded area) of the coordinate system shown in Fig. 3. The abscissa, $\mathscr{R}(s)$, represents the real part of a possible

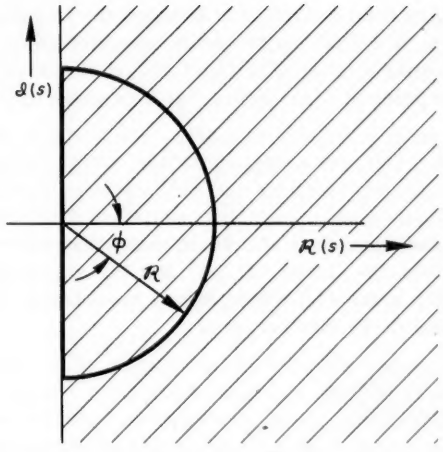

Figure 3—Representation of complex s plane.

root s of Eq. 9, whereas the ordinate, $\mathscr{I}(s)$, represents the imaginary part of the root s. The transfer function $W(s)$ is used in determining whether the characteristic equation has any roots in the right-half plane. From Cauchy's residue theorem, a closed-contour integral involving $W(s)$ along some path C equals the sum of the residues within the closed contour C. Since the residues are associated with the singularities of $W(s)$, and these in turn occur at the zeros of the characteristic equation, a contour integral of $W(s)$ along a path C enclosing the right-half s plane will have a value only if the characteristic equation has a root with a real, positive part.

The function $W(s)$ can be written in the form

$$W(s) \equiv \frac{1}{D(s)} \qquad (12)$$

where $D(s)$ is the characteristic equation. The zeros of $D(s)$ are of concern here, and therefore the contour integral of interest is

$$\frac{1}{2\pi i} \int_C \frac{ds}{D(s)} = \text{sum of residues within } C \qquad (13)$$

where C is indicated in Fig. 3 by ϕ going from $-\pi/2$ to $\pi/2$ and $\mathscr{R}$ is approaching ∞. Rather than dealing with Eq. 13, more information can be obtained by considering the value of

$$\frac{1}{2\pi i} \int_C \frac{D'(s)}{D(s)} \, ds \qquad (14)$$

where $D'(s) = dD(s)/ds$. Although the singularities of $1/D(s)$ occur at the zeros of $D(s)$, the singularities of $D'(s)/D(s)$ occur at the zeros and poles of $D(s)$. Hence, let D be given by

$$D(s) \equiv \frac{Z(s)}{P(s)} \tag{15}$$

where the values of s which make $Z(s) = 0$ constitute the zeros of $D(s)$ and the values of s which make $P(s) = 0$ constitute the poles of $D(s)$. Equation 14 then becomes

$$\frac{1}{2\pi i}\int_C \frac{D'(s)}{D(s)}\,ds = \frac{1}{2\pi i}\int_C \left(\frac{Z'}{Z} - \frac{P'}{P}\right) ds = N_z - N_p \tag{16}$$

where N_z is the number of zeros of $Z(s)$, with each zero counted according to its multiplicity, and N_p is the number of poles of $Z(s)$, with each pole counted according to its multiplicity.

Also, Eq. 14 can be integrated to give

$$\frac{1}{2\pi i}\int_C d\left[\ln D(s)\right] = \frac{1}{2\pi i}\left[\ln|D(s)| + i\theta(s)\right]_C = \frac{\psi}{2\pi} \tag{17}$$

where $|D(s)|$ is the magnitude of the vector associated with the real and imaginary parts of D for a given value of s, measured from the origin; θ is the angle of the vector associated with the real and imaginary parts of D for a given value of s, measured from the real axis; and ψ is the angle swept out in the $D(s)$ plane as s follows contour C. Equations 16 and 17 give

$$N_z = N_p + \frac{\psi}{2\pi} \tag{18}$$

where $\psi/2\pi$ can be interpreted as the number of encirclements of the origin in the $D(s)$ plane as s follows contour C. Thus, by noting the number of poles of $D(s)$, if any, and by determining the number of encirclements of the origin when plotting $D(s)$ as s varies over C, the number of roots of the characteristic equation in the right-half s plane can be obtained (encirclements are considered positive when they occur in a counterclockwise fashion, with s traced in a counterclockwise fashion). The above are essentially Nyquist's stability criteria. Stability is associated with those conditions which lead to no zeros for $D(s)$ in the right-half plane.

Frequency-Response Plots. The path that C follows is the imaginary axis and the extremes of the right-half s plane. Under most circumstances, $D(s)$ is a well-behaved function, and encirclement of the origin is associated only with movement of s along the imaginary axis. Under these circumstances, only the value of $D(i\omega)$ is of concern, where ω takes on all values. Also, when $s = i\omega$, the transfer function $W(s) \equiv 1/D(s)$ can be interpreted as the steady-state response to a sinusoidal input signal having frequency ω; thus this portion of a Nyquist plot is associated with frequency-response data.

The preceding conditions and relations are utilized in the frequency-response plots, or Bode diagrams, which essentially plot the amplitude of the system transfer function $|W(s)|$ and its phase angle as s traverses the imaginary axis. Thus, frequency-response plots are a way of presenting the information contained in the Nyquist plot.

Autocorrelation and Transfer Functions. The autocorrelation function is related to the magnitude of the transfer function $|W(s)|$, where s lies upon the imaginary axis; however, it does not give the associated phase angle. Thus, only part of the information given by a frequency-response plot is obtained by utilizing the autocorrelation function.

Physically, the autocorrelation function correlates the value of a function at time t against the value at a different time. Thus, if $x(t)$ is the variable, the normalized autocorrelation function $\phi(\tau)$ is given by

$$\phi(\tau) \equiv \frac{\overline{x(t)\,x(t+\tau)}}{\overline{x(t)^2}} = \frac{\lim_{T\to\infty}\frac{1}{2T}\int_{-T}^{T} x(t)\,x(t+\tau)\,dt}{\lim_{T\to\infty}\frac{1}{2T}\int_{-T}^{T}\overline{x(t)^2}\,dt} = \frac{\int_{-\infty}^{\infty} x(t)\,x(t+\tau)\,dt}{\int_{-\infty}^{\infty}\overline{x(t)^2}\,dt} \tag{19}$$

Equation 19 implies that the statistical characteristics of $x(t)$ do not depend on the time of observation or that a stationary random process is being investigated. Under such circumstances

the time average is equivalent to an ensemble average, where an ensemble consists of an infinite set of similar mechanisms generating $x(t)$. The integral of interest in Eq. 19 is

$$\int_{-\infty}^{\infty} x(t)\, x(\tau + t)\, dt \tag{20}$$

This is related to the transfer function, as indicated below.

By means of the Fourier integral, $x(t)$ can be written as

$$x(t) = \frac{1}{2\pi} \int_{-\infty}^{\infty} X(\nu)\, e^{-i\nu t}\, d\nu = \mathcal{F}^{-1}\,[X(\nu)] \tag{21}$$

with

$$X(\nu) = \int_{-\infty}^{\infty} x(t)\, e^{i\nu t}\, dt = \mathcal{F}[x(t)] \tag{22}$$

where $\mathcal{F}$ represents the Fourier transform and $\mathcal{F}^{-1}$ represents the inverse Fourier transform. It follows that

$$\begin{aligned}
&\int_{-\infty}^{\infty} x(t)\, x(\tau + t)\, dt \\
&= \int_{-\infty}^{\infty} dt\, x(t) \frac{1}{2\pi} \int_{-\infty}^{\infty} X(\nu)\, e^{-i\nu(\tau+t)}\, d\nu \\
&= \frac{1}{2\pi} \int_{-\infty}^{\infty} X(\nu)\, e^{-i\nu\tau}\, d\nu \int_{-\infty}^{\infty} x(t)\, e^{i(-\nu)t}\, dt \\
&= \mathcal{F}^{-1}[X(\nu)\, X(-\nu)]
\end{aligned} \tag{23}$$

Thus, taking the Fourier transform of $\phi(\tau)$ gives

$$\mathcal{F}\,[\phi(\tau)] = K\, X(\nu)\, X(-\nu) \tag{24}$$

where

$$K^{-1} = \int_{-\infty}^{\infty} \overline{x(t)}^2\, dt$$

The relation between $X(\nu)$ and the transfer function is still required. Since a set of linear equations with constant coefficients is being considered, the kinetic equations can be written in the form

$$D\left(\frac{d}{dt}\right) x(t) = f(t) \tag{25}$$

where $D(d/dt)$ is a polynominal in the differential operator d/dt and $f(t)$ is the input disturbance. Performing a Laplace transform on Eq. 25 gives the transfer function $W(s)$ as

$$W(s) = \frac{\overline{x}(s)}{\overline{f}(s)} = \frac{1}{D(s)} \tag{26}$$

So long as the system is stable, b in Eq. 8 becomes zero, and the Laplace transform essentially reverts to the Fourier transform; thus, for a reactivity disturbance $\delta k_e(t)$ having Fourier transform $\overline{\delta k_e}(\nu)$, $X(\nu)$ becomes

$$X(\nu) = W(\nu)\ \overline{\delta k_e}(\nu) \tag{27}$$

where ν is associated with points along the imaginary axis. Combining Eqs. 24 and 27 relates the Fourier transform of the autocorrelation function to the transfer function. Magnitude alone is given by $\mathcal{F}\,[\phi(\tau)]$; also, $X(\nu)$ and $X(-\nu)$ have the same magnitude since one is the complex conjugate of the other. Thus

$$\mathcal{F}\,[\phi(\tau)] = K|W(\nu)|^2 |\overline{\delta k_e}(\nu)|^2 \tag{28}$$

Assuming the spectral power density of the input reactivity noise to be independent of frequency, $|\overline{\delta k_e}(\nu)|^2$ is a constant, and therefore the magnitude of the square of the transfer function is proportional to the Fourier transform of the autocorrelation function. This remarkable relation permits information to be extracted from a reactor power trace. Although this relation has been used in the past primarily to determine reactor parameters, it may find future use in determining changes in operating conditions.

Application to the HRE-2

During operation of the Homogeneous Reactor Experiment (HRE-2), small power oscillations of 1 to 2 per cent occurred;[7] also, small power excursions of ~3 per cent or greater occurred under conditions associated with loss of fuel from solution. By examining the autocorrelation function associated with the small power oscillations, an estimate[6] of the relative amplitude of the transfer-function frequency response was obtained for various operating conditions. The results indicated that the shape of the resulting curve was dependent upon whether a loss of uranium from the fuel solution had taken place. It is not clear whether sufficiently long time intervals were considered in the time averaging (hand calculations were performed) or what error was introduced by approximating $\phi(\tau)$ by an analytical function. The results indicate, however, that this method may prove to be use-

ful in ascertaining that undesirable operating conditions exist, even though the spectral density of the input reactivity noise may not be independent of frequency. (Paul R. Kasten)

Conference on Transfer-Function Measurements and Reactor Stability Analysis

On May 2 and 3, 1960, a conference convened at ANL on "Transfer-Function Measurements and Reactor Stability Analysis" under the program chairmanship of James A. DeShong, Jr. Detailed minutes of this conference will be issued by ANL. There were four sessions: I, Transfer-Function Use in Reactor Dynamic Studies; II, Thermal Reactor Measurements and Analysis of Stability; III, Fast-Reactor Measurements and Analysis of Stability; and IV, Pile Oscillator Reactivity Measurements; and the first three sessions had direct bearing on reactor safety.

A transfer function describes the output of a system, such as a reactor, as a function of the input. More specifically, for a sinusoidal input of unit amplitude and zero-phase shift, the transfer function gives the amplitude and phase of the output as a function of the input frequency.

J. P. Franz of Westinghouse Atomic Power Division (WAPD) discussed "Reactor Transfer Functions," particularly for a system in which the only nonlinearity is the fact that the rate of change of reactor power (or neutron population) is proportional to the product of reactivity and the already existing power (or neutron) level, whereas the power-to-reactivity feedback is linear. The method suggested for handling such systems involves considering a harmonic reactivity input

$$\rho = \mu \sin \omega t$$

applied to a system in which the transfer function is obtained by linearizing the kinetic equations; the transfer function determines the phase shift. The output of the transfer function is then fed to a "describing" function, which describes the nonlinearity in amplitude. Higher harmonics are neglected, a procedure which was said to be justified if the absolute value of the transfer function decreases with ω, that is, if the reactor's attenuation of the input reactivity increases as the input frequency increases.[8,9] In the discussion, Elias P. Gyftopoulos of MIT and Z. Akcasu of ANL questioned the omission of higher harmonics.

The relative advantages and disadvantages of the transfer-function method of reactor analysis were discussed by Gyftopoulos in a paper entitled "Transfer-Function Representation of Nuclear Power Plant Components." One interesting point referred to the refinements introduced by the nonlinear treatment as compared with the linearized equation. The speaker pointed out that the refinements are important only when the reactivity additions are large, and then distortions of the spatial distribution are usually just as important as the nonlinearities. The transfer functions discussed in the paper were primarily those referring to the reactor itself and, to a lesser extent, those of the other components.

The measurement of the power-to-void transfer function in a SPERT-I channel was described by S. Zivi and R. W. Wright of Ramo-Wooldridge. This function describes the amplitude and phase of the oscillation in steam voids caused by a sinusoidal oscillation (of unit amplitude and zero phase) in the power input. In the measurements, the channel, built to be as similar as possible to the SPERT-I fuel element, was heated electrically, and the voids were measured by means of the transmission of soft X rays through the steam-water mixture in the channel. The transfer function thus measured can then be combined with the other transfer functions of the system, such as the "zero-power" transfer function, which describes reactor power and its dependence on reactivity input, the "coolant flow to void" transfer function, the "void to coolant flow" transfer function, etc. The results are in qualitative but not quantitative agreement with the experimental SPERT-I results, indicating that the actual reactor depends somewhat on phenomena not simulated in the electrically heated model.[10] Descriptions of the instrumentation for "High- and Low-Power SPERT-I Transfer-Function Measurements" and analyses of the results were presented by A. A. Wasserman of Phillips Petroleum Company.

A paper entitled "Combined Reactor and Power System Analysis — BWR" was presented by E. R. Owen of General Electric, San Jose. This paper was particularly timely in view of the just-completed and eminently successful half-power run of the Dresden boiling-water reactor that was built by G. E. Effects discussed

included reactor kinetics, the heating of the fuel, the thermodynamics of the boiling water, the Doppler coefficient, the voids, the flow, etc. Components external to the reactor, such as the 200-ft steam line to the turbine with its compressible flow, were also discussed.

Z. Akcasu and J. A. Thie of ANL presented a paper on "Methods of Investigating Noisy Power Reactors." The noise in a reactor is a handicap if it is intended to measure the transfer function of a reactor with relatively small input oscillations of control rods, feed-water temperature, reflector position, etc. Methods of measurement with high noise-to-signal ratios were discussed. It was also pointed out that noise can be useful as a means of determining the transfer function if correlation between a noisy input and the resulting output is used.

A paper entitled "Measurement and Analysis of the KEWB Transfer Function by Reactor Modulation Techniques" was presented by E. R. Cohen and R. N. Cordy of Atomics International. By theory and by experiment, Cohen and Cordy found that, in a small, reflected reactor which is forced into oscillations of a period comparable to the mean neutron lifetime, the reflector neutrons act like a "seventh group" of delayed neutrons. The ratio l/β, where l is the neutron lifetime and β is the delayed-neutron fraction, was determined on the KEWB up to 260 cycles/sec. The high-frequency transfer functions were different for cadmium-covered and bare neutron-flux indicators.

Stability analyses of the EBWR were discussed by W. C. Lipinski of ANL. Transfer functions were measurable, under ideal conditions, to $\pm\frac{1}{2}$ per cent in amplitude and $\frac{1}{2}$ degree in phase. On the "old" EBWR core, the stability limit was determined to be at 66 Mw, but a slightly larger core that was scheduled for operation in September is expected to be stable at 100 Mw.

J. N. Grace of WAPD presented a paper on "Analysis of a Reactivity Instability Experiment with Boiling." The analysis was performed by using two regions: the lower one nonboiling and the upper one boiling, with the boiling boundary fixed. Variation of the boiling boundary was a second-order effect and could be neglected at smaller amplitudes. In addition, digital calculations on a 30-region model were performed which predicted the observed 5 cycles/sec oscillation. These oscillations turn out to be amplitude-modulated, with a frequency of $\frac{1}{2}$ cycle/sec. In the discussion, Z. Akcasu of ANL suggested that the modulation may result from the situation in which the almost-unstable reactor acts as a narrow band-pass filter excited by its own noise. A second topic by Grace on the precession of the flux around the seed of the PWR because of nonlinear effects was also of interest.

E. S. Beckjord and W. H. Harker of General Electric, San Jose, described "The Dynamics of Large Boiling Reactors," and, in particular, their proof that the xenon flux-tilt oscillations cannot occur unless the reactor operator applies "total power" control, that is, operating all control rods together in order to keep the total power constant. "Local control," that is, the use of control within a given region to keep the power in this region constant, does not result in the spatial oscillation. Local control is employed, for instance, in the Dresden reactor. The proof is based on the assumption that the regions in the reactor are coupled only by neutron flux.

Another paper from WAPD, "Digital Calculations of Spatial Stability Related to PWR," by P. F. Lacy, referred to the spatial oscillations in the PWR seed as a result of xenon instability. The calculations described the observed effect well, but the model used had to be somewhat coarser than desirable to make the time- and space-dependent calculation possible.

The papers of the third session all referred to fast reactors and were particularly timely in this country because of the construction of the Enrico Fermi Atomic Power Plant, which will go into operation in the near future. As is well known,[11] the EBR-I, an experimental fast-breeder reactor, exhibited a positive power coefficient and had an accident during an experiment, after a long period of satisfactory operation. The seriousness of a positive power coefficient in a fast reactor is aggravated by the short mean neutron lifetime.

The power coefficient in pressurized-water and boiling-water reactors[12] depends largely on expansion of the moderator and void formation, and these effects are, in principle, generally well defined by the type of the reactor. The power coefficient in the fast reactor was traced, as described in great detail in the papers presented, to mechanical deformation of the fuel elements as a function of reactor power: A fuel element in a flux gradient will get hotter on the side in the higher flux; it will expand more on

that side and will tend to bow, with the convex side of the bend toward the higher flux. If this bowing makes the reactor more compact, a positive power coefficient results. This was the case with EBR-I. In this reactor the elements were fastened at the top and bottom.

Other methods of fastening the elements can result in negative power coefficients or in substantially lower positive power coefficients. The fact that the bowing is responsible for the tendency toward instability was demonstrated by constraining the elements by means of ribs, which gave a nuclearly very stable reactor; then some of the ribs were sheared off, and the tendency to instability, as exhibited by the behavior of the transfer functions, returned. The matter is somewhat complicated by the fact that the constraints may allow "limited bowing" at small amplitude but will prevent large deflections. Also, if the experimental transfer functions were extrapolated to zero frequency, one would expect to find the static temperature and power coefficients; however, this was not the case, probably because of very slow effects such as vessel expansion or time delays in the secondary system. These effects were too slow to be detected, even with the lowest frequencies used in the transfer-function measurements.

The general conclusions of the speakers in the third session were that (1) the positive power coefficients are understood in principle and can be dealt with, and (2) the fast reactor can be constructed with predictable nuclear stability. The papers presented were as follows: "Oscillator Tests in British Fast Reactors," A. R. Baker, UKAEA, Harwell; "Progress in Frequency-Response Measurements at Los Alamos," R. M. Kiehn, Los Alamos Scientific Laboratory; "Recent EBR-I Stability Studies," R. R. Smith, ANL; "Analysis of the Power Feedback Relations in the Enrico Fermi Reactor," F. Storrer, Atomic Power Development Associates; and "Kinetics Studies on EBR-I and EBR-II," D. Okrent, ANL.

(W. K. Ergen)

Argonne Transient Reactor Test Facility (TREAT)

The acronym TREAT refers to a homogeneous uranium-graphite reactor that has been designed to test fast-reactor fuel elements to meltdown during a self-limiting fast transient. The core is protected from damage during the excursion by its relatively high heat capacity, and the excursion is controlled by the negative temperature coefficient of the core. The need for this kind of test information is established by noting that the consequences of core meltdown comprise the largest set of unknown factors in evaluating fast-reactor safety.

The physics analysis of the TREAT design[13] was summarized in the June 1960 issue of *Nuclear Safety*.[14] This facility has also been discussed in a paper presented at the second Geneva Conference,[15] a hazards summary report,[16] a postcriticality study,[17] and several papers[18–22] presented at American Nuclear Society meetings. Initial results of meltdown tests on EBR-II fuel pins were described in the June 1960 issue of *Nucleonics*.[23] Some of the design parameters are indicated in the following:

Nominal core size, ft	$4 \times 5 \times 5$
Reactor dimensions, ft	$8 \times 10\frac{1}{3} \times 10\frac{1}{3}$
Carbon-to-U^{235} ratio	10,000
Excess reactivity with no slots or holes	0.19
Control-rod worth (central 8 rods)	0.28
Average integrated neutron flux during excursion, neutrons/cm^2	4×10^{15}
Final average graphite temperature, °C	235
$-\delta k_{eff}$ in temperature coefficient, %	4.2 to 5.9
Minimum reactor prompt period (self-limiting), msec	30 to 40
Peak flux, neutrons/(cm^2)(sec)	$\sim 10^{17}$
Neutron lifetime, sec	8.55×10^{-4}
Steady-state power (maximum), kw	100
Maximum transient energy release, Mw-sec	1000

The first set of experiments[18] performed on the reactor yielded a neutron lifetime value of 8.8×10^{-4} sec and a temperature coefficient of 1.8×10^{-4} over a range of 20°C. Other measurements of the temperature coefficient[19] gave results as high as 2.1×10^{-4}. Some of the observations made during these first experiments are listed below:

Criticality increment, %	Peak power, Mw	Remarks
0.4	2.5	Subprompt critical
0.6	8.1	Subprompt critical
0.8	43.0	Superprompt critical

The power vs. time curve develops a narrowing symmetrical peak as the criticality increment

increases in value, reflecting the diminishing effect of the delayed neutrons.

A two-stage program[22] of meltdown experiments has been developed to obtain needed data on the characteristics of the reactor and to pave the way to full-scale testing of fast-reactor fuel elements. These experiments will be carried out with single, uncooled, unirradiated EBR-II Mark I type fuel pins. Some of the first results of these single-element meltdown tests are summarized as follows:

Maximum temperature, °C	Condition
530	No damage
675	No damage
820	Void in bond
860	Void in bond
887	Void in bond
945	Helical void in bond
950	Penetration of cladding by uranium
960	Helical void in bond
965	Helical void in bond
980	Void in bond
1000	Eutectic coating on fuel; no penetration of cladding
1000	Uranium melted through bottom; no evidence of violent disintegration
1015	Vigorous penetration
≥1030	Violent disintegration; molten uranium ejected from element

The EBR-II fuel specimens tested[23] consisted of uranium castings containing 5 per cent "fissium" alloy in concentric stainless-steel tubes. The alloy and cladding were bonded with sodium, and the containing capsule was filled with helium at 1 atm. The enrichments ranged from normal to 6 per cent U^{235}, which is lower than the 49 per cent to be used in EBR-II, a fast reactor, but the enrichment is ample for TREAT, a thermal reactor. Experiments with fuel-rod clusters are being planned. It is the opinion of the reviewer that the experiments performed thus far confirm the original design concepts. (P. H. Pitkanen)

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Section III

REACTOR SAFETY FEATURES

Safety System Interconnections Between Experiment and Reactor

When the first test reactors were designed, much thought and attention were given to the layouts and logical designs of the safety systems that would shut down the reactors on the occurrence of any dangerous situation within the core. It was not realized until much later that large-scale experiments to be performed within a reactor had potentially dangerous conditions associated with them and that the control and safety systems required for the experimental facilities were, in many cases, more complex, more expensive, and more prone to trouble than the control and safety systems associated with the reactor. Since most test reactors can have several experiments operating simultaneously, the resulting complexity of experiment controls and the failures normally associated with such equipment can give rise to an intolerable amount of trouble from malfunction and false operation of the experiment control and safety systems.

Handbooks[1–3] have been prepared to guide experimenters at four of the large, presently operating, test reactors. These reactors are the Materials Testing Reactor (MTR) and the Engineering Test Reactor (ETR) at the National Reactor Testing Station in Idaho and the Low Intensity Test Reactor (LITR) and the Oak Ridge National Laboratory Research Reactor (ORR) at Oak Ridge, Tenn. The systems described in the handbooks are quite similar; a typical system, the one in use at the ORR and the LITR, is shown diagrammatically in Fig. 4. For each experiment a decision must be made as to which process variables, if any, must be used as bases for shutting down the reactor if their values show potentially dangerous conditions. For each such variable, it must then be determined what the time response of the necessary reactor shutdown must be. In each of these reactors, more than one mode is available by which to shut down the reactor. These modes vary in the speed with which the reactor is shut down and also in the ease with which full power may be recovered after the reactor shutdown signal is cleared, either because the process variable no longer is in the dangerous range or because the signal was false to begin with.

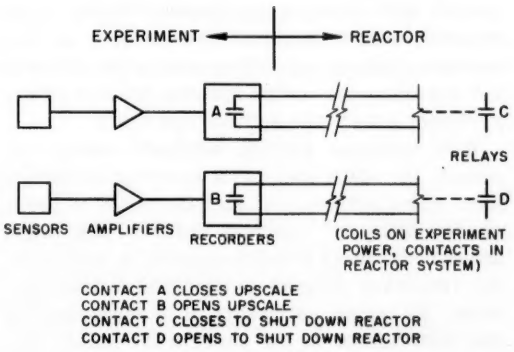

Figure 4 — Typical double-track reactor shutdown from an in-pile experiment. (The system shown is the type used at the ORR and the LITR.)

It is apparent from studying the handbooks[1–3] that there is a significant history of false shutdowns from reactor experiments. A very high premium is therefore placed upon use of one of the less drastic modes of reactor shutdown in cases where the time response of this alternate mode is adequate. In rare cases, a single process variable will have associated with it more than one mode of reactor shutdown, but usually the extra false shutdowns generated in the extra equipment more than compensate for

the possible added advantage of having the less drastic action available. For each mode of reactor shutdown, a double-track electrical signaling system interconnects the experiment control and safety system with the reactor control and safety system. This is done so that a single failure in this vital interconnection cannot paralyze the necessary action.

In the MTR and ETR systems, both the paths in the double-track link from the experiment to the reactor control system are provided with relays that close in the event of trouble, but they are arranged so that they will not give signals upon the loss of power to the experiment control system. In this event, a separate monitor ensures that the reactor will be shut down when all control power to the experiment instrumentation is lost. This is done to avoid a false shutdown from momentary dropout of the relays involved. On the other hand, in the LITR and ORR systems the two tracks in the link have opposite aspects, as shown in Fig. 4. Energizing one circuit or deenergizing the other circuit will transmit the desired signal to the reactor control system. This is done so that common failures of both tracks in the link will not paralyze the entire system; the loss of experiment power will activate one track.

Both systems provide adequate safety, although the ORR and LITR systems appear to have some advantage in simplicity and in lower susceptibility to common failures that might affect both tracks simultaneously. In any event, the two-track concept is essential to both systems. To ensure reliability, it is necessary for the instruments in the experiment safety system, as well as those in the interconnection, to be double-tracked so that the correct operation of the system will never be dependent upon a single component. It is vital to the system that the two channels of information and control be truly independent so that they will not be prone to a common failure whereby both channels are inoperative for the same reason. Although double-tracked channels will provide a greater degree of assurance in obtaining a safety action when needed, the probability of false safety actions is also increased. For this reason, careful examination of the number of parameters that can cause reactor shutdown is essential in order to reduce unnecessary shutdowns to a minimum. It is also clear that a controller must not be used for any safety function since the failure of such an instrument will change the variable which it controls, in many cases toward the dangerous values, at the same time that the safety function is paralyzed because of failure of the same instrument.

These precepts of redundancy and independence may seem self-evident now, but it is apparent from references 1 to 3 that they have been developed with some difficulty. Standard procedures and accepted good practice have been described in considerable detail so that new experiments may be designed to conform in these respects to those already in progress, and the maintenance and operating personnel need not be confronted with a variety of different schemes. It is clear that both of the organizations responsible for the operation of these test reactors place a high premium on strict conformance to established practice in order to ensure smooth operation of their facilities without impairment of the necessary safety.

(S. H. Hanauer)

A Pneumatic Temperature Probe for Gas-Cooled Reactors

Pneumatic probes for the determination of gas temperatures were described by Schmick[4] and Cassan[5] as early as 1929. The major developmental work did not take place, however, until after World War II and the advent of the jet engine. The pneumatic temperature probe, also referred to as the thermodynamic or total-temperature probe, is a rather novel system of measuring the temperature of a high-velocity high-temperature gas. In this service, it has many advantages over thermocouples. The pneumatic probe is not affected by radiation or convection; it measures total temperature, including that resulting from velocity; it samples a small, well-defined area; and, with proper cooling, the range can be extended to above 4000°R.

The design operating temperature of a gas-cooled reactor is, in general, limited by the safe operating temperature of the fuel elements.[6] In order to obtain optimum performance of the reactor, it is therefore desirable to operate all the fuel channels as close to the design temperature as possible. The safest way to do this is to measure either the fuel-element temperature or the fuel-channel-outlet gas temperature and to regulate the gas flow through each channel to give the desired fuel-element

temperature. Since, at the temperatures of interest, thermocouples do not have adequate reliability over the expected core life, it appears that the pneumatic temperature probe is well matched to the needs of a gas-cooled reactor.

The basic advantages of the pneumatic temperature probe have made it a very useful instrument for obtaining jet-engine exhaust temperature profiles and other measurements of importance to jet-engine evaluation.[7–11] A pneumatic probe was used recently in England to evaluate the performance of open-hearth furnaces in the steel industry.[12,13]

The pneumatic temperature probe is, simply, a tube which has two flow restrictions (see Fig. 5). Hot gas is passed through the first

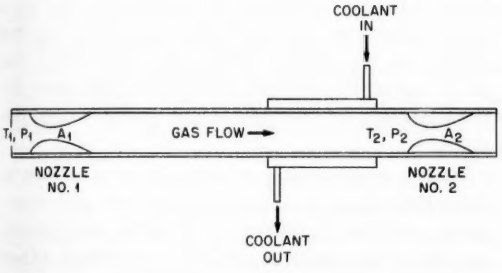

Figure 5—Typical pneumatic temperature probe.

restriction and then cooled. The second restriction is used to measure the mass flow rate through the first. When both the pressure and the mass flow rate are known at a restriction, the temperature can be determined through the use of the flow equation. A simplified system can be obtained by designing the restrictions so that the ratio of the throat pressure to the inlet pressure is less than the critical value for the gas used. This will produce "sonic flow" in the throat and eliminate the necessity for measuring the differential pressure at each restriction since the flow rate under these conditions will be dependent on upstream pressure alone. A further advantage can be gained by using a convergent-divergent nozzle as the restriction so that the coefficient of discharge will not change as the pressure ratio is decreased below the critical value. This type of restriction will also give good pressure recovery, usually of about 80 per cent.

The mass flow rate, W, for a critical flow nozzle is given by the equation

$$W = \frac{PA}{(T)^{1/2}} \left(\frac{2}{R}\right)^{1/2} \left\{\frac{\gamma}{\gamma - 1}\left[\left(\frac{P_t}{P_i}\right)^{2/\gamma} - \left(\frac{P_t}{P_i}\right)^{(\gamma+1)/\gamma}\right]\right\}^{1/2} \quad (1)$$

where A is the nozzle area, P is the inlet pressure, T is the inlet temperature, R is the gas constant, γ is the ratio of specific heats (C_p/C_v), and P_t/P_i is the critical pressure ratio. Subscripts i and t refer to inlet and throat pressures. Equating the mass flow at each nozzle and solving for T_1 (refer to Fig. 5) gives

$$T_1 = T_2 \left(\frac{C_1 A_1 P_1 F_1}{C_2 A_2 P_2 F_2}\right)^2 \quad (2)$$

where C is the coefficient of discharge and

$$F_1 = \left\{\frac{\gamma_1}{\gamma_1 - 1}\left[\left(\frac{P_{t_1}}{P_{i_1}}\right)^{2/\gamma_1} - \left(\frac{P_{t_1}}{P_{i_1}}\right)^{(\gamma_1+1)/\gamma_1}\right]\right\}^{1/2}$$

Equation 2 can be further simplified by grouping the constants:

$$T_1 = KT_2(P_1/P_2)^2$$

where K, the calibration constant, is

$$K = \left(\frac{C_1 A_1 F_1}{C_2 A_2 F_2}\right)^2$$

If T_2 is controlled at a fixed value, which can be done with a small heat exchanger, the measured temperature becomes a function of a pressure ratio.

Typical probes are from ½ to 1 in. in diameter, with nozzles 0.1 to 0.030 in. in diameter at their throats. The geometry of the nozzle is not critical. A quadrant of a circle or ellipse serves very well as the approach section, as will a hyperbolic curve. The throat should be one throat diameter down from the entrance. For the divergent section, either a straight or a slightly curved section works equally well. The nozzle should be constructed of a hard material so that changes in flow characteristics caused by erosion will be minimized. Because of the many factors appearing in the equation as squared terms, very careful application of the device is necessary if reasonable accuracy is to be obtained. Large temperature differences between the first nozzle and the gas will result in heat transfer to the gas and will cause nonisentropic expansion. Another source of error would be a change in mass flow rate after the first nozzle.

This could be the result of a leak, a chemical reaction in the gas, or condensation in the cooled tube. Thermal expansion of the nozzle will result in errors of several per cent if it is not considered. As an indication of the tolerances required, an over-all accuracy of 2 per cent would require that (1) the pressure ratio be measured to within 0.75 per cent, (2) T_2 be controlled to within 0.5 per cent, and (3) the calibration constant be known to within 0.75 per cent. The calibration constant will vary if the physical properties of the gas change or if the geometry of the nozzles is altered. A simple method of calibration can be achieved by making T_1 and T_2 the same temperature, as would be the case if both were at room temperature. From the equation it can be seen that, if T_1 is equal to T_2, the value of the calibration constant will be equal to the square of the pressure ratio.

In view of the apparent suitability of the pneumatic temperature probe for the determination of temperatures in a gas-cooled reactor, it is now intended to provide such a measurement on the outlet of all channels from the core of the Experimental Gas-Cooled Reactor (EGCR). The pneumatic temperature system in the EGCR will utilize the piping for the fuel-cladding-rupture detection system.

The rupture detection system in the EGCR will have a 0.4-in.-OD stainless-steel tube running from the outlet of each fuel channel to a valving manifold. There will be a nozzle installed in each of these tubes near the fuel channel which will serve as the entrance nozzle for temperature measurement. The valving manifold will sample two channels simultaneously. One sample will be routed to a radiation detector for rupture detection, and the second sample will be taken through a heat exchanger and a second nozzle to complete the pneumatic temperature system. The upper plenum pressure will be taken as P_1, and the pressure P_2 will be measured just before the second nozzle. Since the temperature of the gas entering the second nozzle is to be controlled, the fuel channel exit gas temperature will be a function of only the pressure ratio P_1/P_2. Because of the experimental nature of the pneumatic system, conventional thermocouples will be installed in each channel to serve as a backup and to provide a means of evaluating the pneumatic temperature probe as a gas-cooled-reactor temperature monitor. (T. L. McLean)

Electric Power Systems of Nuclear Power Plants

The electric power system of a nuclear power plant is important both for the distribution to the external grid of the power produced in the plant and for the continued functioning of essential equipment in the facility in the event of a failure in any part of the system. These two functions establish a requirement for reliability of the electrical connections to, and the distribution system within, a nuclear power plant which is seldom matched in industry.

There is an economic incentive for delivering uninterrupted power to the consumer, and therefore at least two connections to the electrical grid are required in order to minimize outages from external failures. Furthermore, the continued operation of a nuclear plant following the failure of any single electrical bus within the plant requires multiple buses and judicious diversification of the electrical loads within the plant.

Power must also be available at all times to essential equipment (such as shutdown and emergency cooling systems, ventilation equipment, and filtering systems), which is provided either to prevent the occurrence of serious accidents or to minimize the consequences thereof. The need for dependable sources of power for instruments and control systems is even more obvious since the loss of this power would necessitate costly shutdowns, if not serious accidents. Thus, in addition to one or more external power lines, most nuclear plants must include one or more sources of emergency power, such as batteries and diesel-driven generators.

Although dependability of the power sources is a requirement common to all types of nuclear power plants, the type and complexity of the electric power systems may vary to a great extent from one facility to another. In particular, the urgency of the need for emergency cooling influences the electric system designs of the various plants. An attempt is made here to describe the electric power systems of presently operating and currently planned nuclear power plants and to evaluate their reliability. No attempt has been made to evaluate the power systems with respect to the presumed requirements of the various plants. The descriptions of the electric systems were taken from hazards re-

ports. It must be emphasized, however, that most of the hazards reports from which data were extracted are preliminary. In many cases the description of the system is incomplete, and there may be significant changes in the designs before the facilities are placed in operation.

The principal features of the various electric systems are presented in Table III-1, and each system is discussed in this review, except the Saxton Reactor Facility, the Small Pressurized-Water Reactor, and the Vallecitos Boiling-Water Reactor. The available information on the electric power systems for these plants was insufficient to permit comparable discussion.

Shippingport Atomic Power Station (PWR)[14]

The primary power supply to the PWR consists of two 138-kv transmission lines originating at the Phillips generating station of the Duquesne Light Company and one 23-kv transmission line running to a system junction with the Phillips generating station and the power system of the Pennsylvania Power Company. The 23-kv line is normally open and can be used to provide emergency power by manual switching. The two primary lines occupy essentially separate rights-of-way and form a closed loop with the Shippingport Atomic Power Station as a junction point. The junction consists of a solid 138-kv bus from which one feeder is connected to a 138/2.4/2.4-kv three-winding transformer, and another feeder is connected to a 138/15.5-kv transformer that has its low-voltage side connected to the turbine-generator, as well as to a 15.5/2.4/2.4-kv three-winding transformer. The four 2.4-kv windings of the three-winding transformers feed into four individual buses. Normally open tie breakers permit the conversion of the setup to a two-bus arrangement if required. Each of the four main coolant pumps is connected to a separate 2.4-kv bus. All other power needed for the reactor and auxiliary equipment emanates from the same four 2.4-kv buses in pairs of feeders connected to transformers arranged to provide a bus system arrangement similar to that for the primary 2.4-kv sources.

The unique features of this electric system arrangement are the use of a solid primary bus and the consistent pattern applied for all subsystem distribution. The loss of one 138-kv transmission line does not require any automatic or manual switching, and the prevailing conditions of plant operation are not altered in any way because the remaining transmission line and the plant turbine-generator provide uninterrupted power for all plant requirements. The loss of both 138-kv transmission lines could be alleviated by manually closing the 23-kv emergency transmission line, which can be fed from the Pennsylvania Power Company system in the event that the Duquesne Light Company system becomes completely dead. Automatic transfer of this line is not required because there is ample time available before damage can occur to vital equipment. A diesel-driven generator is available to provide emergency power in case of complete plant isolation from the outside power systems. Here again, only manual start-up of the diesel-driven generator is required. A second diesel-driven generator might be desirable as an additional precaution, but the past history of the tie lines is good, and the probability of a simultaneous failure of the tie lines and the diesel-driven generator is small.

The reactor control system makes use of 120-volt a-c power, with source transfer circuitry in case of a failure, and the safety system is fed from a 125-volt battery that is backed up by an emergency battery. The turbine-generator plant also uses 125-volt d-c power from its own battery, which is backed up by the same emergency battery as that for the safety system.

Consolidated Edison Thorium Reactor (Indian Point)[15]

Power for the auxiliary equipment of the Indian Point station is supplied from four lines, each of which feeds into one of the four sections of the 13.8-kv auxiliary bus. Three of these lines originate at a three-section 138-kv bus at a nearby substation; normally closed automatic tie breakers permit sectionalizing of the bus as needed. The fourth line is connected to the plant generator. Switching facilities are provided for interconnections of the 13.8-kv bus sections. Three 138-kv transmission lines feed the high-voltage bus sections. Two of these lines interconnect with a common remote junction, and the third connects with an independent power system. Power for control of the reactor and for operating its safety system is derived from three battery-operated motor-alternators, one of which is equipped with an additional steam turbine drive. Two batteries are provided, and the power supplies for the battery chargers and

Table III-1 ELECTRIC POWER SYSTEMS OF VARIOUS NUCLEAR POWER PLANTS*

Short name of plant	External primary power	Line rights-of-way	Primary bus arrangement	Plant power generation, Mw	No. of generators	Plant auxiliary power distribution feeders
PWR	Two 138-kv lines	Separate, closed loop	Solid	68 gross 60 net [100 gross]	1	Four 2400 volts; four 480 volts for reactor; four 480 volts for turbine-generator; two 480 volts for screen house
Indian Point	Three 138-kv lines	Common, radial	Sectionalized, normally closed automatic tie breaker	275 gross 255 net	1	Four 13.8 kv and multiple 440 volts
Yankee	Two 115-kv lines	Separate, closed loops	Solid	145 gross 136 net	1	Three 2400 volts, three 440 volts
N.S. *Savannah*	None; plant primary power supplied by 1500-kv turbine-generators		Sectionalized, normally closed automatic tie breaker			
Saxton	[Four 115-kv lines]	[Three separate]		[5.75 gross] [5.0 net]	[Existing]	
SPWR				16.5 net	1	
VBWR				5	1	
Dresden	Five 138-kv lines	Separate, closed loops	Sectionalized, normally closed tie breaker	192 gross 180 net	1	Two 4.1 kv, four 480 volts
Elk River	[Four 69-kv lines]	[Separate]	Solid	22	1	Two 440 volts (one standby)
Pathfinder	Two 115-kv lines	Separate, closed loop	Solid	66 gross 62 net	1	Two 2.4 kv
Humboldt Bay	External lines not indicated; two existing conventional units on common bus			60 gross	1	Two 2.4 kv (one standby)
BONUS	Two 38-kv lines		Solid	17.3	1	One 480 volts
Big Rock Point			Solid	75 gross	1	Two 2.4 kv (one standby)
Hallam				75 gross	1	Two 2.4 kv (one standby)
Enrico Fermi	Two 120-kv lines		Sectionalized	104 gross 94 net	1	Three 4.8 kv (one standby)
EGCR	One 13.8-kv line	Radial	Solid	25	1	Two 2.4 kv, four 440 volts
FWCNP	Two 65-kv lines		Solid	50	1	Two 2.4 kv (one standby), two 600 volts
Peach Bottom	One 33-kv line and [one 220-kv line]	[Separate, radial]		40	1	[480 volts]
Piqua	Two 4.1-kv lines	Common		11.4	Existing	
CVTR	Four 114-kv lines; one 66-kv line	Separate, closed loop	Two solid	30	Existing	Two feeders from existing steam plant

*Entries in brackets represent changes made by the reactor designer since the publication of the hazards reports used for this comparison. Since the reviewer has no additional information explaining such changes, these changes are only listed for additional information.

Table III-1 (Continued)

additional information explaining such changes, these changes are only listed for additional information.

Table III-1 (Continued)

Short name of plant	Auxiliary bus arrangement	Plant emergency source	Reactor control power	Reactor control emergency power
PWR	Four sections for reactor, normally open tie breaker; four sections for steam generators, normally open tie breaker	One 500-kw diesel, manual; one 23-kv line	120-volt a-c 125-volt d-c (safety system)	125-volt battery, manual transfer
Indian Point	Four sections, normally open tie breaker	None except as derived from external primary power lines	120-volt a-c alternators	
Yankee	Three sections, normally open tie breaker	One diesel, manual	120-volt a-c alternators	One diesel
N.S. *Savannah*	Two sections, normally closed tie breaker	Two 750-kw diesels, automatic; one 300-kw diesel, automatic start, manual switching	Two 120-volt a-c alternators (one standby) One 120-volt, a-c from auxiliary transformer	120-volt a-c from auxiliary transformer
Saxton		[Existing steam plant with two turbine-generators]		[120-volt a-c from station emergency source]
SPWR				
VBWR				
Dresden	Two sections, normally open tie breaker	One 500-kw diesel, automatic; one 34.4-kv line	125-volt d-c battery; two 120-volt a-c alternators (safety system)	
Elk River	Solid	One 35-kw diesel, automatic	120-volt a-c alternator; d-c for safety system	120-volt a-c from station source
Pathfinder	[Two sections, each with one normal and one emergency 2.4-kv feeder]	[One 125-kv diesel]	[One a-c alternator driven by battery-power motor]	
Humboldt Bay	Solid	[Bus from existing units; one diesel]	Two 120-volt a-c alternators	
BONUS	Solid	One 100-kw diesel	120-volt a-c	120-volt a-c standby feeder
Big Rock Point	Solid		Two 120-volt a-c alternators with gasoline engine as emergency drive	
Hallam	Solid	One 400-kw diesel, automatic	Two 120-volt a-c alternators	120-volt a-c station feeder
Enrico Fermi	Solid	One 80-kva diesel	120-volt a-c alternator; 125-volt d-c	
EGCR	Two sections, normally closed tie breaker for 2.4-kv feeders; two 2-section buses, normally open tie breaker for 440-volt feeders	Two 1000-kw diesels, automatic	Two 120-volt a-c alternators; two batteries	
FWCNP	Two sections for 2.4-kv feeders; two sections, normally open tie breakers for 600-volt feeders	One 200-kw diesel	Two 250-volt batteries	
Peach Bottom		Diesel	[Critical power from a-c system]	[Battery]
Piqua		One 250-kw diesel, automatic	One 120-volt a-c alternator; two batteries on same bus	120-volt a-c station feeder
CVTR	Two sections, normally open tie breaker	One feeder from hydro plant		

essential loads are well diversified. Circuit-breaker control is from the batteries.

This system permits reliable diversification of essential loads. It is noted, however, that the feeder lines and some of the transmission lines use common rights-of-way. Furthermore, there seems to be no provision for emergency power for the decay-heat cooling pumps in the event all external primary power sources fail.

Yankee Nuclear Power Station (Yankee)[16]

Two 115-kv transmission lines and the plant generator connect to a common high-voltage bus at the Yankee plant. A feeder from each incoming line and from the plant generator energizes the three-section main distribution bus, whose normally open tie breakers may be closed manually. The 480-volt distribution bus also contains three sections that are fed from feeders originating at the three incoming 2400-volt lines. Their tie breakers also are normally open and may be closed manually. The main characteristic of this arrangement for power distribution is the temporary reduction in reactor power level upon loss of one transmission line until a manual transfer of vital equipment to another bus section can be made. A diesel-driven generator provides power in case of a complete loss of primary auxiliary power.

Vital control power is obtained from a battery-driven dc-ac alternator, with an alternate emergency source from the 480-volt plant system. Two 125-volt batteries provide d-c power for station requirements.

Only one dc-ac alternator is provided, with transfer to the emergency feeder being manual. The reviewer questions the reliability of this system for continuous power production, since there would be a temporary interruption of vital control power during transfer to the emergency power supply. Any interruption of control power presently results in immediate scram of the reactor. Alternative arrangements are being evaluated in order to provide improved operational reliability.

Nuclear Merchant Ship Reactor (N.S. Savannah)[17]

The N.S. *Savannah*, being mobile, must provide all its own auxiliary power for safe operation. Power is normally supplied by two turbine-generators feeding into a two-section bus with a normally closed tie breaker. The loads are diversified among a number of load centers, each center being connected to both main auxiliary bus sections through an automatic transfer arrangement. Two diesel-driven generators are automatically connected to the main auxiliary bus sections in case of unavailability of the turbine-generators. Also, an emergency diesel-driven generator located on the navigation deck is connected to an independent emergency bus (which is provided with switching facilities to the main auxiliary buses), as well as to load centers that supply the primary coolant pumps, pressurizer heaters, and other critical components inside the containment vessel.

The essential reactor components are energized through automatic transfer switching circuits to permit their operation from either bus section. Since one turbine-generator has insufficient capacity for the normal electrical load, provision is made for automatic starting and paralleling of the two 750-kw diesel generators. In keeping with marine practice of multiple alternative means of power supply and distribution, the plant can operate with one bus section available, although reduction in reactor power is necessary. If automatic transfers are needed, it would be more reliable to run the system with the bus tie breaker open, provided that the separate system frequencies can be tolerated and that each turbine-generator can carry all the load.

The critical loads, such as control, safety, and nuclear instrumentation, are connected to a 120-volt a-c bus that is connected to two dc-ac alternator sets backed up by a 125-volt battery. One alternator is a standby. The battery floats on its bus, which is energized by two ac-dc motor-generator sets, each with its feeders connected to separate 450-volt a-c bus sections.

Since one of the dc-ac motor-generator sets is termed a standby, the failure of the set in operation would temporarily deenergize the safety system and shut down the reactor. Parallel operation of the two sets would be preferable, and subsequent information states that this is to be the normal mode of generation.

Dresden Nuclear Power Station (Dresden)[18]

Auxiliary power for the Dresden plant is furnished by two transformers, one connected to one section of a two-section 138-kv bus and the other connected to the station generator leads. The generator in turn is connected to

the second section of the 138-kv bus via a transformer. Primary power to the high-voltage bus, which has a normally closed tie breaker, is provided by five transmission lines, two of which follow one right-of-way and the other three follow another right-of-way. Two 4.1-kv load centers are selectively connected to the two auxiliary power transformers, and all the essential loads of either center are automatically transferred to the other in the event either transformer fails. If both the transformers were to fail or if all the 138-kv transmission lines were to fail, a standby 34.5-kv emergency line would be automatically connected to one of the main load centers, and, at the same time, an emergency diesel-driven generator would be automatically started and kept in readiness. The diesel-driven generator would automatically be connected to the 480-volt system if the emergency line should also fail. The 480-volt system contains two load centers, each having two main buses with a tie breaker. Each load-center bus section is connected through its own transformer to the 4.1-kv bus sections. The whole system should be capable of supplying reliable auxiliary power at all times, the only concern being the reliability of the automatic transfer equipment and the consequences of its failure to perform as expected.

Power for critical control and safety circuits is available from two ac-ac alternators connected to two normally separated distribution buses. Failure of one alternator would necessitate manual transfer of the affected bus to a station source. A 125-volt battery is provided for station requirements. The system is well designed for assuring continuous power, and, in conjunction with the station battery, it should provide the necessary reliability for operation and control.

Elk River Reactor (Elk River)[19]

The 13.8-kv plant generator at the Elk River plant is connected through a transformer to a solid 69-kv bus. This bus also supplies a 69-kv/480-volt plant reserve transformer. Although not indicated on the reference material, the bus is further firmed up by two 11.5-Mw steam-driven generators. Auxiliary power is normally furnished by a 13.8-kv/480-volt transformer connected directly to the plant generator and is distributed from a 440-volt station service bus through two main switchgears. Power is fed from this station service bus through a single circuit breaker to a tie point from which the circuit splits into two branches, each of which is connected to one of the two switchgears through a separate breaker. The circuit that connects the main auxiliary power transformer to the tie point is a feeder section common to both switchgears. In the event of failure of the normal auxiliary supply, the two main switchgears are transferred to the plant reserve transformer through a switching arrangement identical to that of the main auxiliary supply. Loss of both supplies is backed up by the automatic starting of a diesel-driven generator. The system contains several features that the reviewer considers to be undesirable:

1. The transfer from one auxiliary source to the other involves a temporary loss of all auxiliary power, possibly a complete loss, if the transfer should fail.
2. A failure in the common feeder circuit requires the transfer of *all* loads to the other source.

With the indicated redundancy of external power sources, a more reliable system from the 69-kv bus down could have been accomplished, and the need for automatic transfer and dual purpose feeder sections could have been avoided.

Power to essential instrument and reactor control circuits is supplied from a battery-driven dc-ac alternator set. An automatic throw-over to a station supply is provided in case of alternator shutdown or failure.

In the most recent hazards report, Final Hazards Report for the Elk River Reactor (July 8, 1960), the results of power failure are described.

In case of loss of power from both electrical services, the superheater auxiliaries will trip, resulting in a four-rod scram placing the reactor on hot standby. Since the battery power supply provides power to control-rod clutches and instruments, the remaining rods will not scram. In a few seconds after the power interruption, the engine generator will provide power to Motor Control Center No. 3, which supplies power to all essential reactor auxiliaries, such as the shield cooling pumps, purification system pumps, air monitor vacuum pump, and control-rod motors.

Depending on the cause of the power interruption, the reactor can be shut down, at the

operator's discretion, by inserting the remaining rods.

Pathfinder Atomic Power Plant (Pathfinder)[20]

Auxiliary power is provided at the Pathfinder plant by a stepdown transformer from the solid 115-kv bus and also by a stepdown transformer connected to the 13.8-kv generator leads. The generator output is stepped up and connected to the high-voltage bus. Two transmission lines, each with independent rights-of-way, connect the plant with the Northern States Power Company system. A bypass and disconnecting switch is provided, apparently to assure emergency power in case of a 115-kv bus fault. The referenced document[20] does not indicate the existence of any other power sources. Internal emergency supplies would be required in the event of total primary power failure, an occurrence which must be considered possible. Were the bypass switch mentioned to be an oil circuit breaker, it would save time in switching. The reviewer believes that the reliability of the auxiliary power sources should be improved by providing an additional independent source. No information was presented on the provisions for assuring power for the reactor control system.

Humboldt Bay Power Plant (Humboldt Bay)[21]

The arrangement of the bus structure and the number of incoming lines for the Humboldt Bay plant have not been documented, although in any event power is available from two existing conventional units at the same site. Auxiliary power is normally obtained from the station generator, which feeds into a solid bus.

An emergency diesel generator that supplies a 480-volt bus is furnished to supply power to those auxiliaries required for safe shutdown of the plant. Loss of voltage from the 480-volt bus supplying these auxiliaries automatically transfers the load to a 480-volt bus being supplied from either of the two existing generator units, and simultaneously the emergency diesel generator is started and kept in readiness. If this bus cannot supply the load or is not energized, then the load is transferred to the already running diesel generator.

A preferable a-c system, composed of two separate buses and two ac-dc motor generators, is used to supply power for reactor instrumentation, safety circuits, and annunciators. A station battery provides the d-c power required.

Boiling Nuclear Superheater Reactor (BONUS)[22]

At the BONUS plant, two 115-kv transmission lines connect to the primary bus system, which consists of main and transfer buses into which the station generator feeds through its transformer. From this system, two service transformers, with both transformers operating in parallel, supply auxiliary power to a solid 4.1-kv bus. From this bus, two feeders with transformers that also operate in parallel are connected to one section of a double-ended bus whose tie breaker is normally closed. The other bus section provides the power for the essential shutdown loads and can be connected to an emergency diesel-driven generator. Operating the feeders in parallel increases the reliability of the system. It is understood that electrical storms are rather severe in the area in which this reactor is located and that such storms could disrupt both incoming lines simultaneously, and therefore the internal plant emergency power reliability assumes greater importance.

A battery furnishes the required d-c power. The reactor control and safety system is operated from a 120-volt a-c bus that is normally connected to the 4.1-kv bus through a single transformer. A standby feeder and transformer from the 480-volt system is automatically connected to the 120-volt bus in the event of failure of the normal feeder. This arrangement is believed to be undesirable with regard to the continuous production of power for the following reasons: (1) control power should not be lost at any time; (2) there is only one on-line source of control power so that it will scram the reactor in the event of the failure or temporary interruption of that power source.

High Power Density Reactor Plant (Big Rock Point)[23]

The auxiliary and control power system for the Big Rock Point plant is identical to that of the Humboldt Bay Power Station, described previously.

Hallam Nuclear Power Facility (Hallam)[24]

External power sources have not yet been described in the hazards reports for the Hallam plant. The solid 4.1-kv auxiliary power distribution bus is normally supplied from a station

transformer with an alternate normally open feeder from a reserve station transformer, as an emergency source, through an automatic transfer breaker. Three 480/277-volt switchgear sections are connected to the main bus through individual transformers. Two of these sections have a common tie breaker that is normally open and will automatically close if either of the transformers associated with the two sections fails as a source. A diesel-driven generator is automatically connected to one of these two sections in the event both transformers fail as supplies. The reviewer believes that the main weakness of this arrangement is that too much reliance is placed on the ability of automatic transfer equipment to perform with reproducible time delays and actions. It is likely that at some time the transfer might not take place or that it might provide enough time delay to permit all contactors to drop open. In either event the supply for critical loads would be lost, and the reactor would be scrammed unnecessarily. Another weakness is that a bus fault on the 480-volt section associated with the diesel-driven generator could make this emergency power unit unavailable when needed.

A single 480-volt feeder supplies the vital control center from which three battery chargers (one a standby) connect to two separate 250-volt d-c buses. A floating station battery and a dc-ac alternator set are associated with each of the d-c buses. The adequacy of only one feeder for the vital distribution center is questionable.

Enrico Fermi Atomic Power Plant (Enrico Fermi)[25]

Two incoming 120-kv transmission lines connect to the two-section high-voltage bus at the Enrico Fermi plant, one line to a section. The normal position of the tie breaker is not indicated. The solid 4.8-kv reactor plant auxiliary bus is provided with two sources from the primary system and one source from the station generator circuit. The latter is normally in standby. Four 480-volt distribution buses are provided for the various areas of the reactor portion of the plant. Each bus is connected to a transformer, and two are paired on the 4.8-kv side through a common circuit breaker that connects to the reactor auxiliary bus. The four 480-volt distribution buses can be joined in pairs through normally open tie breakers. The reviewer feels that this will be a good system if the detailed design is properly executed. There are, however, two points of concern: (1) the two incoming power lines follow parallel rights-of-way separated by 200 ft and may be subject to a common failure, and (2) the use of a common breaker for two transformers significantly increases the equipment affected by a single failure. The absence of automatic transfer schemes is a step in the right direction. Complete loss of primary power is covered by a diesel-driven generator that is connected to the 480-volt station system. The reactor auxiliary bus can be bypassed for maintenance purposes.

Control and instrument power is 120-volt a-c from a dc-ac alternator set with a tie to the station power. The safety-system power is also 120-volt a-c from two dc-ac alternators connected to isolated buses and to separate batteries. The safety-rod system uses d-c power. The whole system appears to be reliable and reflects sound engineering.

Experimental Gas-Cooled Reactor (EGCR)[6]

The 13.8-kv, solid, main-station power bus is connected to the station generator and to one 13.8-kv outside feeder at the EGCR. Two transformers supply a two-section 2.4-kv bus, with a normally closed tie breaker, from the main-station bus. Two double-ended load centers, one for normal loads and the other for emergency loads, with normally open tie breakers are connected (each through its own transformer) to the 2.4-kv bus. The reviewer believes that the addition of at least one or more independent outside power sources would enhance the reliability of this system in the event of a complete primary power failure. The emergency source required after a scram is provided by two diesel-driven generators that are automatically switched to the distribution buses of the emergency load center.

According to the hazards report,[26] critical control power is obtained from two dc-ac alternators, the d-c sides being connected to separate 125-volt batteries and the a-c sides to individual bus sections with a normally open tie breaker. Battery chargers normally carry the load, and their supply originates at the two sections of the emergency load center. This system does not represent the final design, and therefore its evaluation is postponed.

Florida West Coast Nuclear Power Plant (FWCNP)[26]

Two incoming 69-kv transmission lines at the FWCNP plant connect to a solid primary bus to which are tied the generator transformer and the auxiliary power transformer, which is normally in operation. A second auxiliary power transformer can be connected to either of the two transmission lines by means of motor-operated disconnects. The low-voltage sides of these two transformers are tied to two solid 2.4-kv bus sections, but only one transformer supplies power during normal operation. These two buses are then effectively run as a solid bus by means of two normally closed feeder breakers from the same auxiliary transformer. In case of failure of the first transformer, transfer to the second transformer is automatic. The 600-volt auxiliaries are connected to two buses with a normally open tie breaker, each bus section being fed from a transformer supplied by one of the 2.4-kv buses. A diesel-driven generator and two 250-volt batteries provide power in case of loss of primary power. The automatic transfer designed into this plant could cause momentary interruptions of power to vital equipment. Also, a failure of the auxiliary transformer that is normally in operation would compel the operator to shut down the generator and disconnect the 69-kv lines. Since an auxiliary transformer is already available to ensure continuation of auxiliary power, it would appear that slight modification would permit maintaining uninterrupted generator output.

All control power is provided by 250-volt batteries. Although the information concerning control power is incomplete, the concept of direct current as the control source is noteworthy.

Peach Bottom Atomic Power Station (Peach Bottom)*[27]

The hazards report for the Peach Bottom plant lists the available sources of power; however, the power source arrangement and method of distribution are not given. The single 220-kv line connecting the station generator to the system is a solid tap to a high capacity system intertie. Plant control will be designed so that the main turbine-generator will continue to supply auxiliary power upon loss of the incoming line. However, the main turbine-generator cannot be counted on as a power source in an emergency, and there is no assurance that the generator will stay on the line during a transient created by the loss of the incoming line. Backup power, adequate for orderly plant shutdown, will be supplied by a 33-kv power line originating at the Conowingo Hydro Station. An emergency diesel-driven generator will supply power for safe shutdown in the unlikely event of loss of all other power sources or other unusual conditions. A station battery will provide reliable power for reactor control and essential plant control and instrumentation.

Piqua Organic-Moderated Reactor (Piqua)[28]

The primary incoming power for the Piqua plant consists of two 4.1-kv lines that connect to a solid station bus. To this bus are connected the two main pumps and a single auxiliary transformer which supplies all other station loads. Failure of this transformer would drop all station loads connected to it, which appears to the reviewer to constitute a rather inadequate arrangement. A diesel-driven generator is provided to supply emergency power; however, there is no need to resort to emergency power so long as the two primary sources are capable of delivering the necessary power. (An interesting feature of this plant is that the steam produced is piped to existing turbine-generators at Piqua; it has no station generator itself.)

Control power is obtained from a solid bus with two batteries in parallel. One dc-ac alternator with a flywheel provides noise-free instrument power to a distribution panel, which can be connected to a station feeder in case the alternator fails. The provision of only one alternator raises the question as to how the concept of coincident circuitry can be carried out even though coincident circuitry is provided in the safety system itself. [In general, at least three supplies are considered optimum for coincident circuitry to accomplish the two major aims: (1) prevent unnecessary shutdowns from false signals and (2) allow testing of safety channels to proceed with plant in full operation. With only two supplies, if one fails, it is difficult to determine whether the other is intact.]

*The description of this facility, as described in the Site Evaluation Report,[27] was modified, as described in a letter dated Sept. 12, 1960, from S. R. Gaty, Vice President of Philadelphia Electric Company, to the editor.

Carolinas-Virginia Tube Reactor (CVTR)[29]

The CVTR is integrated with a steam plant and a hydroelectric plant at the same site. The primary system comprises four 114-kv transmission lines, and a fifth line connects the steam plant with the hydroelectric plant. Auxiliary power for the reactor plant is distributed at 480 volts from a two-section bus with a normally open tie breaker. Each section connects to a feeder from 13.2-kv bus sections in the steam plant. An emergency supply line is provided and connected to the hydroelectric plant system. Auxiliary power reliability is further increased through switching facilities to several existing steam generators. The control power provisions are not documented.

Conclusions

Based upon an analysis and comparisons of the collective hazards summary reports, and the engineering judgment of the reviewer, the following conclusions have been evolved which hypothesize the optimum power supply system for a nuclear power plant.

The reliability of the auxiliary power for a reactor plant is only as good as the reliability of the primary power sources provided for that plant. In many instances, it appears that the plant site selection has a considerable influence upon the extent to which the designer can go in obtaining multiple external power sources. For continued operation, at least two outside sources must be available so that the loss of one source will not require shutdown or a decrease in the level of power output. Furthermore, these outside power sources should run along divergent rights-of-way to minimize the frequency of simultaneous interruptions during severe storms (instances of which have been personally observed by the reviewer). The latter requirement has not been met by some of the installations, probably because of site selection and financial considerations.

In the majority of the plants examined, a complete failure of the primary supply is offset by providing one or more diesel-driven generators that are automatically started and tied to the essential postscram loads. Other plants rely on the availability of an emergency line from a different system before resorting to the use of the available diesel-driven generator. In other cases, no indication is given as to the existence of any internal independent emergency supplies. Some designs apparently attribute to the station generator the same source reliability as that of the incoming lines. However, there can be no positive assurance that the generator will continue to operate after a complete severance from the outside system. Furthermore, during a reactor shutdown, the station generator is not available at all as a source. In the case of the CVTR station,[30] the problem of emergency power is simple because the nuclear plant is integrated with existing steam and hydroelectric plants at the same site. A similar advantage will be obtained when nuclear plants accommodate two or more reactors. Some reactors allow a considerable lapse of time upon complete power loss before damage to certain components can be expected. Even so, an independent emergency source in the plant is the best safeguard against the most extreme eventualities.

The design treatment for the distribution of the auxiliary power follows various concepts. Some designs represent an adaptation from conventional plant methods, and other designs indicate an earnest endeavor to integrate the peculiarities of a reactor plant into a system that will provide acceptable reliability. As in the case of the primary transmission lines, two or more secondary feeders, connecting to either a solid bus or sectionalized buses, operating as a solid bus (with their tie breakers normally closed) allow the loss of one source without recourse to automatic transfers or reduction in power level. The presence of a tie breaker permits isolation of a bus fault; however, the reliability of a properly designed solid bus is probably as high as or higher than the reliability of automatic switching equipment. Where two or more bus sections are used with normally open tie breakers, load diversification assures at least partial load operation in the event of failure of one of the multiple sources until full-source supply can be restored by manual or automatic transfers to the operable sources. With large reactors the tendency seems to favor multiple centers with broad load diversification and interconnecting facilities for the many bus sections. A temporary partial loss of load is unavoidable with such arrangements in the event of loss of an auxiliary feeder. This drawback can be avoided through the installation of dual full-capacity components, such as circulating pumps that operate in parallel and each pump connected to a different source of power. This

arrangement certainly has great merit, even though the capital investment is increased.

In order to satisfy the primary criterion of reliability, the control power source must never fail. Continuous control power must be supplied even in the event of failure of one source, and thus two or more sources arranged to provide uninterrupted service to the control system are necessary. In several of the installation hazards reports reviewed, the designs presented do not meet these requirements.

Most of the installations depend on some form of coincidence circuitry in their safety systems as a means of avoiding unnecessary shutdowns; however, the proper application of these circuits demands consistency throughout, from power supply to sensors. It is useless to apply coincidence at the sensors and not at the power supplies. Most plants depend on dc-ac alternator sets in suitable arrangements to provide power to the "source-sensitive" nuclear instruments. Few plants use d-c power for the safety circuits. Batteries are inherently more reliable than rotary equipment, and the use of solid-state devices operating on d-c sources would allow a-c sources to be abandoned in favor of batteries. This simplification of source requirements should result in greater reliability of the safety systems.

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Section IV

PLANT SAFETY FEATURES

Explosions and Fires in Compressed-Air Systems

By JOSEPH J. DiNUNNO*

Several instances of explosions and fires in compressed-air systems have focused attention upon the hazards associated with the inadvertent entry of oil into air supplies and on the prevailing design practices for assuring the safety of compressed-air installations. Three explosions of air-oil mixtures occurred at nuclear reactor installations during the fall of 1959, none of which resulted in a nuclear incident. Two of the explosions occurred at the National Reactor Testing Station (NRTS) in the aftercoolers of the similar, large air compressors at the MTR and ETR installations. The third explosion occurred in the air flask of a hydraulic system on the USS *Triton* prototype at West Milton, N. Y.

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Summary of Recent Incidents

The compressors involved in the two NRTS incidents[1] were lubricated with a petroleum-base oil which had a flash point in the range of 440°F and which had been recommended by the compressor manufacturer for this service. The first occurrence, i.e., the explosion at the ETR, involved a 4000-hp motor-driven three-stage compressor designed to deliver air at a pressure of 300 psig. The first indication of trouble came from annunciator alarm signals that indicated high temperatures of the crankcase-oil cooler and the effluent water from the aftercooler. Almost simultaneously a loud noise was heard, and smoking of the aftercooler shell was observed. The second explosion, i.e., the one at the MTR, involved a 1500-hp motor-driven three-stage compressor that was also designed to deliver air at a pressure of 300 psig. Trouble was recognized when an operator's helper at the compressor heard a loud unusual noise and then observed that the paint was blistering and smoking along the entire length of the shell of the aftercooler.

In both these incidents, the fuel for the fire that followed the explosions was deposits of compressor lubricating oil in the aftercooler. Although the exact cause of ignition was not determined, the following events and postulated mechanisms of ignition were considered:

1. Water-flow failure in the intercoolers between the first and second compressor stages could have resulted in temperatures high enough to ignite carbon, scale, and oil deposits, which in turn could have ignited oil vapor in the air stream.

2. Spontaneous ignition could have resulted from air under pressure passing over a mixture of oil and rust. It was noted that the final-

stage temperatures had gone above 320°F at a 300-psig discharge pressure.

3. Failure of an aftercooler tube could have allowed the compressed air to force the cooling water out of the compressor. The resulting localized heating could have been sufficient to cause ignition.

4. Any of the following factors (such as static sparks in the air stream, oil breakdown, or the formation of peroxides or carbon-oxygen complexes) could have caused ignition of the oil vapor.

The corrective measures taken after these two incidents included (1) conversion to a fire-resistant synthetic lubricant, (2) modification of piston rings to prevent intermixing of the crankcase oil and the synthetic lubricant, (3) addition of alarms on cooling-water lines to warn of flow shutoff, (4) addition of a closed cooling-water loop at one plant on a trial basis for improved control of temperature differentials across cylinders, (5) installation of manual blowdown valves on interstage piping, and (6) installation of traps on separators and aftercoolers. In order to assure proper diffusion of the lubricant and thus prevent its accumulation in the system, changes were made so that the compressors would run constantly at full capacity, and the excess air available under partial load conditions would be dumped.

The third incident,[2,3] which was of a somewhat different nature, occurred on the USS *Triton* prototype at West Milton, N. Y., on Oct. 30, 1959; and, as in the NRTS incidents, the explosion did not involve the nuclear portion of the plant. An air-oil explosion, followed by a self-extinguishing flash fire, occurred in an air flask component of a 3000-psi hydraulic oil system that normally served to actuate valves in the circulating-water and steam systems of the propulsion plant. The air flask was used to pressurize an accumulator consisting of a cylinder containing a free-floating piston that separated the air from the hydraulic oil. The experts who surveyed the situation following the explosion agreed that the flask showed no evidence of prior metallurgical defect and that oil could and had entered the high-pressure air flask, thereby creating a potentially explosive mixture. In this case also, the mechanism by which the explosion was initiated was not definitely established, but the following events and postulated mechanisms of ignition were given as possible causes:

1. Lifting of an air-relief valve could have started a shock wave of sufficient intensity to ignite an air-oil mixture in the air flask.

2. Structural failure of the flask could have released an explosive mixture of oil and air. This mixture could have been ignited by hot metallic particles resulting from the bursting of the flask.

3. Spontaneous combustion could have resulted from decomposition or other chemical reaction of the oil with the high-pressure air in the flask.

4. Ignition could have been caused by compression or shock waves created by rapid opening of a valve in the high-pressure system.

Corrective action in the USS *Triton* prototype case included (1) replacement of the oil in both the hydraulic system and the air compressor with a fire-resistant synthetic lubricant, (2) use of nitrogen in lieu of air to charge the flask, (3) improvement in the design of the accumulator piston to provide double-seal protection and interseal venting, (4) provisions for complete drainage of the air flasks, and (5) the addition of a missile shield around the hydraulic plant.

Review of Reactor Installations

As a followup of these fire and explosion incidents, a review of the application of compressed-air systems in reactor plants was initiated. The review was directed toward establishment of the following:

1. Whether high-pressure air-oil systems similar to that of the USS *Triton* prototype existed in other AEC or licensed reactor plants

2. What hazards are associated with the use of compressed-air systems

3. What design features would eliminate or minimize dangers from air-oil fires or explosions in compressed-air systems

Reviews of plant descriptions in the hazards reports on major reactor plants, such as Dresden, Yankee, VBWR, N.S. *Savannah*, and Shippingport, showed that none of these plants use high-pressure air-oil systems of the type that exploded on the USS *Triton* prototype. The N.S. *Savannah* has an oil-pressurized accumulator as part of the control-rod scram system, but the pressurizing medium is nitrogen. The

Shippingport reactor has a high-pressure valve-operating flask charged with air, but the hydraulic fluid is water. Low-pressure systems (100 to 200 psi) are being used for rod-drive control schemes in some cases, and they are used almost universally for instrument and valve control functions. Low-pressure air for control of reactor plant components is being supplied most commonly from the nonlubricated (carbon-ring) types of air compressor. In several instances where this was not the case, followup action has led to a changeover to fire-resistant lubricants for the compressors. General-service low-pressure air for auxiliary systems in such plants is supplied from petroleum-base oil-lubricated compressors. These oil-lubricated compressor installations include such safeguards as aftercoolers, oil filters, and air-temperature monitors.

The reports[1,2] on the USS *Triton* prototype and on the ETR and MTR experiences were disseminated to AEC operations offices, and a request was issued for a survey of systems in AEC installations which might involve mixtures of air and oil under high pressure. Reports from the field offices indicated no widespread application of high-pressure air-oil systems of the type that led to the USS *Triton* prototype incident. Results of the dissemination of the incident reports led to improvements in systems in several instances and created greater general awareness of hazards associated with high-pressure air-oil applications. However, the reports from the field offices showed differences in the emphasis given to low-pressure air systems using oil-lubricated compressors. At some installations where low-pressure air systems were used, the existing safeguards were evaluated; whereas, in other instances, it was merely assumed that no hazards existed in the application of low-pressure systems. The ETR and MTR experiences with oil-lubricated compressors and the published information on hazards associated with compressed-air systems do not, as indicated below, support the view that low-pressure systems supplied from oil-lubricated compressors are inherently safe.

Recommended Safeguards for Compressed-Air Systems

Industrial experience was examined to determine what hazards might be involved in using compressed-air systems in general, and, in particular, the low-pressure systems in common use in reactor plants. This study revealed that fires and explosions have occurred in compressed-air systems since the earliest days of the air compressor and indicated that this subject requires serious attention with respect to reactor plant design.

The numerous articles that have been published on the phenomena and mechanisms of fires and explosions in compressed-air systems reveal that such incidents have not been confined to high-pressure systems. The common factor in all incidents was that oil from the compressor was in the air supply. The mechanisms and phenomena[4,5] associated with air-oil fires and explosions may be summarized as follows:

1. A fraction of the compressor lubricating oil can be entrained in the delivery air and deposited in the form of fine droplets in the piping, air coolers, and receivers. Oxidation of the oil results as a function of the air temperatures and pressures involved. Under appropriate conditions of temperature and oil-vapor concentration in the air, the vapor ignites and causes an explosion or fire. These conditions are most likely to occur when the oxidation reaction has been accelerated through rapid air flow and/or high temperatures and the air supply is suddenly cut off.

2. Once a flammable gaseous mixture has formed in a pipe of sufficient length and diameter, the deflagration may initiate an explosion that is fed directly by the oil deposit and is capable of propagating at high speed.

3. The formation of a carbon-oxygen complex on carbon deposits from air compressors has been experimentally observed.[6] The formation of such a complex is an exothermic reaction.

4. Chemical decomposition of oils to form peroxides can result from long-term contact of high-pressure air with oil, and the small amounts of sulfur sometimes present in lubricating oils may create sulfur gases that are flammable.

The high-temperature conditions that must accompany the formation of explosive air-oil mixtures in order for fires or explosions to occur can be caused by the following:

1. Oil-impregnated rust or carbonaceous deposits over which high-velocity air flow may

heat up and give off flammable vapors that can ignite spontaneously.

2. Where an initial explosion is sufficiently severe to initiate a shock wave, the temperature within the shock front can be sufficient to vaporize and ignite oil deposits in a pipe.

3. The chemical action in the formation of carbon-oxygen complexes is exothermic and could act as a source of ignition.

4. Ignition could result from compression or shock waves caused by rapid opening of a valve in a high-pressure air system where oil is present.

5. Failure of cooling-water supplies to either the compressor or aftercoolers could result in increases in air temperatures to above the flash point of the lubricating oils.

6. Leaking discharge valves in the compressor could result in abnormal air temperatures that would increase to above the flash point of the lubricating oils.

7. High-temperature intake air or clogged intake filters could cause excessive temperatures of the delivered air.

8. In the case of air-starting lines, engine backfire could cause glowing carbon to blow past check valves and ignite flammable vapors in air receivers and piping.

9. Some oils break down to form carbonaceous deposits on piston and discharge valves, and high-velocity air flow over such deposits could cause excessive heating.

The need for special attention to the safety aspects of compressor installations and operations has been repeatedly expressed in the literature. For example, in 1937, Gibbs discussed the safety aspects of both the layout of a compressor and its operation in the *Compressed Air Magazine*.[7] Additional support for such safe practices in compressor layout and operation was given by Brown in *Oil Weekly*[8] in May 1943. A further discussion of safe practices appears in an article by Attaway in *Petroleum Refiner*.[9] Inasmuch as Attaway's summary is comprehensive, the practices he has listed are given below:

1. The proper amount of oil to be used in the air cylinder depends upon the viscosity of the oil. Little oil is required to maintain operation. Lubrication of the air cylinder, when incorrectly handled, is believed to be one of the largest causes of explosions. At a pressure as low as 100 lb, the temperature of an air cylinder often reaches 480°F, which is sufficient to vaporize certain grades of lubricating oil. When this vapor is mixed with air, an explosive mixture is formed which may be readily ignited by a glowing bit of carbon or by temperature alone. A good rule suggested for oil lubrication was 1 to 4 drops per minute for small compressors, 6 drops per minute for medium sizes, and 10 to 12 drops per minute for large compressors.

2. Only high-grade oil that contains no animal or vegetable oil should be used. Even the best oil decomposes to some extent and leaves a carbon deposit that is carried into the receiver and piping. This carbon is flammable and soon becomes oil-soaked. Since it is under air pressure and exposed to great heat, it may glow or burn, thus igniting an explosive mixture of oil vapor. The carbon deposit may be distributed to form a dust which may explode when mixed with air and ignited.

3. Faulty compressor discharge valves cause a sharp increase in air temperature. Friction alone may ignite carbon. Compressor valves should be cleaned at least once each month, or oftener if service is continuous.

4. Explosions may occur by ignition of flammable vapors drawn in at the intake. The intake should be located where air is clean and cool. An air filter should also be installed.

5. The air intake pipe should be 8 to 10 ft above ground level, or above the roof, yet as short and straight as possible to avoid pockets, turns, or traps where substances may accumulate. If the intake has such pockets, drains should be installed and checked daily. Moisture has no effect on fire or explosions, but it does cause a corrosion effect.

6. Every means should be used to keep the cylinder temperature down. This will decrease vaporization and decomposition of lubricating oils. This is accomplished by keeping ample cooling water circulating, drawing only clean cool air into the cylinder, and operating the compressor at a speed no greater than rated.

7. Deposits of scale form on the cylinder water jacket and reduce the cooling effect. It is advisable to clean these jackets at least once a year.

8. Each air compressor should have a fusible plug in the discharge piping to relieve pressure when temperature exceeds a certain limit—perhaps fixed at 350 to 500°F, depending upon the compressor size.

9. Discharge piping should be at least the diameter of the opening in the air cylinder and should be carried full size to the receiver. Bends, turns, and loops must be avoided[2] . . . .

10. The receiver should be as close as possible to the compressor and in a cool place. Each receiver should have a pressure gauge, safety valve, and blowoff cock.

11. A shutoff valve should never be placed between the compressor and the receiver unless a

safety valve is placed on the compressor side of the valve. Air outlet from the receiver should be at the top.

12. All piping, fittings, and equipment must withstand a test pressure of 1½ times the normal operating pressure. Air receivers should be constructed to ASME Code.

13. Air starting lines should have two check valves located at least 10 pipe diameters apart. In case of engine backfire, glowing carbon may be blown past a single check and ignite flammable vapors present in the air receivers and piping.

The use of fire-resistant lubricants is usually given as a means for minimizing fires and explosions in compressed-air and hydraulic systems. Such fluids exist in various forms, including synthetic lubricants, mixtures of water and water-soluble synthetic lubricants, and emulsions of petroleum oils and water obtained with the use of emulsifying agents.[10–17] Although such fluids appear to have many desirable features that enhance the safety of the equipment with which they are used, they are not without certain limitations that should be recognized. The aqueous-base fluids are often referred to as "smuffer" type fluids, which impart fire resistance through release of a protective steam blanket in the presence of high temperatures. Since such fluids depend on water content for fire resistance, their application requires a temperature limit (generally of the order of 150°F) and a periodic makeup of the water content. The viscosity of such a fluid changes, of course, with changes in the water content. Other special considerations in the use of aqueous-base fluids include the effects on paints. Softening and the lifting of many of the conventional types of paint and coating will result from the use of aqueous-base fluids. The high vapor pressure of the fluid at elevated temperatures, caused by its high water content, will result in loss of water and some of the volatile inhibitors. Zinc- and cadmium-containing alloys or coatings could not be used in such applications because they are frequently dissolved by aqueous fluids. Grease is not soluble in these fluids, and excess lubrication might result in filter clogging.

Nonaqueous fluids of the phosphate ester type might harm packings compatible with petroleum oils. Static seals might swell in time, and such swelling of dynamic seals would cause them to tighten and rub on moving parts. If a system is being changed over to utilize this type of fire-resistant fluid, a change of seals is recommended.[16] It is understood that butyl type rubber is resistant to such fluids, but a softening effect occurs on Neoprene or Buna N rubber.

Hydraulic Accumulators

Since an accumulator was involved in the USS *Triton* prototype incident, a brief review was made of safe practices in accumulator design. Various accumulator designs exist which include gravity, spring-loaded, pneumatic, inflexible separator, and flexible separator types.[18]

Manufacturers of hydraulic accumulators have recognized standards[19] covering the safety design aspects of such components; these standards were adopted at a Joint Industry Conference in 1959. The following criteria are quoted from the standards:

1. The accumulator shall be constructed to withstand at least five times the operating pressure of the hydraulic system it serves.

2. Means shall be provided for safely relieving accumulator gas and liquid pressure before the accumulator can be disassembled.

3. Hydraulic circuits incorporating accumulators shall be arranged so that the system can be bled at the high point of the circuit.

4. Hydraulic circuits incorporating accumulators shall be interlocked to vent or isolate accumulator fluid pressure when the power is shut off.

5. In circuit application, where accumulator pressure is isolated, full information shall be given on or near the accumulator for proper servicing to prevent injury to personnel.

6. Gas accumulators operating above 200-psi charging pressure shall be charged with nitrogen unless fire-resistant fluids are used, in which case air is permissible.

Conclusions

The incidents covered by this report, together with the published information on this subject, show that the potential hazards of systems involving compressed-air and oil mixtures are indeed real and that such hazards are not peculiar to high-pressure systems alone. Furthermore, adequate design practices have been developed over the years to enhance the safety of such systems, and these design practices should not be compromised or ignored.

instead, the application of safe design practices, coupled with recognition by operating personnel of the potential hazards involved and the need for thorough preventive maintenance, should do much to minimize the possibility of incidents similar to those experienced at the USS *Triton* prototype and at the MTR and ETR installations. (Joseph J. DiNunno)

Pressure-Suppression Containment

Nuclear power plant designers have always been concerned with the nonproductive costs imposed upon the reactor portion of the plant in order to achieve "adequate" safety. Most imposing, in both cost and size, is the containment vessel surrounding the nuclear portion of most plants to contain radioactivity in the event of a major reactor accident. Of the several new approaches to containment which give promise of increased economy, one of the more interesting approaches for boiling- or pressurized-water reactor systems is that known as steam-suppression containment. Beck[20] has described this scheme in the following terms:

> The principle of [the] steam-suppression containment scheme is to locate the reactor and its primary system so that the steam which might emerge from any break in the system is transmitted through a tank of water before it can be released into the building enclosing the facility. As the steam traverses the water path, it is condensed into water, hence no steam pressure is built up in the building, and most of the fission products, except noble gases, would also be retained in the water. The external containment vessel would then need to have only nominal pressure capacity, and leakage of noble gases only need be considered.

Before any containment scheme can be regarded as acceptable, its performance must, of course, be demonstrated to the satisfaction of the AEC. The lack of experience with containment systems, other than the proven pressure vessels, and the expense of ascertaining their performance data are the principal factors that have inhibited their use in the nuclear power plants constructed to date. Of the many power reactors proposed for construction in this country, steam-suppression containment is planned for only one, the Humboldt Bay Power Plant of the Pacific Gas and Electric Company. In their application for a construction permit,[21] the Pacific Gas and Electric Company proposes to build a nuclear power station employing a 200-Mw(t) boiling-water reactor provided with pressure-suppression containment. Although the construction permit for this facility has not yet been approved, the Company is continuing with a development program intended to demonstrate the sufficiency of such containment for the Humboldt Bay plant (see Section VI, p. 111).

The Humboldt Bay reactor vessel contains 1790 cu ft of water and is enclosed in a 12,000-cu ft vessel called the dry well. The dry well is vented through a set of pipes that terminate in a 20,000-cu ft pool of water. This pool, along with a 32,375-cu ft gas space above it, is enclosed by a third vessel, which is called the suppression chamber. The dry well, because it is vented through a water seal, is designed only for the maximum transient pressure increase during a release. The suppression chamber is designed to take only the air and other noncondensable gases from the dry well because the steam is to be condensed in the pool. The radioactive contamination would presumably not escape from the suppression chamber and would probably be stopped during the steam condensation in the pool. Since the rest of the structure, which houses the turbine, condensers, and other auxiliary equipment, does not need to be pressuretight, the cost will be substantially less than the cost of a pressuretight containment vessel.

The utility of a containment scheme such as this is quite dependent on the reactor type, which, in turn, can be characterized by the maximum credible accident. For the Humboldt Bay reactor the maximum credible accident is defined as the worst coolant-loss accident that could result from near-instantaneous complete severance of any pipe connected to the reactor vessel. Such an accident, incidentally, would include (1) the possibility that the broken pipes or other failed parts will become missiles, (2) blast effects, (3) reactor-vessel reaction from the escaping steam and water, and (4) leakage of fission products into the steam. Included in the events that were not considered credible was the rupture of the reactor vessel. In order to justify this assumption, the Pacific Gas and Electric Company used the very impressive statistic that, in over 4000 boiler-years of

operation at pressures in excess of 600 psia, there has never been a vessel failure.

The sequence of events during the maximum credible accident would be as follows: an 8-in. feed-water line to the reactor would rupture and allow the superheated water to pour from the reactor vessel into the dry well and flash into steam. After the water had been expelled, the steam from the reactor vessel would then follow. Because of the higher mass flow rate of water, the feed line failure would result in a higher peak pressure in the dry well than the failure of the 12-in. steam line. Air and steam would exhaust from the dry well through the pool into the suppression chamber. At operating conditions the available thermodynamic energy is about 36 million Btu, and the residual nuclear energy (120 sec) available is 4 million Btu. The vents from the dry well originate as six 40-in.-diameter pipes, each of which manifolds its flow into eight 14-in.-diameter nozzles directed vertically into the pool, with a submergence of 6 ft. The transient pressure in the dry well from the release of water vapor and liquid from the pressure vessel and the venting of the vapor through the vents will reach about 23 psig. The 20,000 cu ft of water in the pool will rise in temperature 36°F, and the pressure in the suppression chamber, because of the addition of air from the dry well, will rise to 9 psig. When the core goes dry because of loss of coolant, a backup core cooling system will function so that the fission products will be essentially retained at their source in the fuel. Nevertheless, the sudden release of pressure and the mechanical action of the steam and water rushing out of the primary system may cause some fuel-rod failures.

Of the questions that arise concerning the feasibility of a scheme such as this, perhaps the most important is whether the escaping steam can be condensed in the pool at a rate fast enough to prevent excessive dry-well pressures. In order to answer this question convincingly, extensive calculations were made on an assortment of models, and three sets of experiments were undertaken. In the first of the experiments, performed at the Moss Landing Power Plant of Pacific Gas and Electric Company, the behavior of a jet of steam in water was examined. A large tank of water, compartmented to simulate the suppression-chamber pool, was used to condense steam from 4- to 14-in.-ID jets at steam rates from 10,000 to 93,000 lb/hr and at submergences from 1 in. to 6 ft. The results of over 100 runs in this facility established unequivocally that pure steam will condense at a very high rate in water. Some instabilities were noted when the water temperature rose to as high as 150°F, and some escaping steam was noted when the submergence was reduced to only a few inches. Normally, the length of the jet from the nozzle was on the order of 2 ft or less.

The second set of experiments was performed with a model scaled approximately to one-tenth of the linear dimension of the reactor. The system consisted of an electrically heated pressure vessel, which was separated from the dry well by a rupture disk; a changeable orifice to restrict the flow; a set of jets; a pool; and a suppression chamber, which in the model was a vessel of sufficient strength to assure safety. During the 30 runs performed in this facility, the transient pressure rise in the dry well followed the calculated prediction with reasonable accuracy. The suppression-chamber pressures were also similar to the predicted values; however, the facility was not so accurate to the scale in regard to the suppression chamber.

The third set of experiments is in progress and is intended to clarify some points of weakness in the original experiments and to employ some details of the system design which were evolved since the original experiments were undertaken. Specifically, a full-scale segment of the suppression chamber (1:6) is being mocked up which will run with pure steam and steam-air mixtures to establish that (1) there will be no oscillations in pool level which could result in the uncovering of a jet and (2) the presence of various amounts of noncondensable gas in the steam will not affect the rate of condensation sufficiently to allow the buildup of steam in the suppression chamber to intolerable pressures.

The pressure-suppression concept is a bold step toward the simplification of boiling-water-reactor containment and is being carefully scrutinized by the Advisory Committee on Reactor Safeguards and by the AEC. If it is found to be acceptable, the Humboldt Bay Power Plant containment scheme will probably replace the conventional pressure dome for many future reactors. (M. E. Whatley)

Safety in Hot-Laboratory Design and Operation

The "hot" laboratory, i.e., a laboratory designed for remote handling of radioactive and toxic materials, is an integrated system of safety devices for protecting personnel from the radioactivity and the toxicity of some of the materials handled therein while, at the same time, permitting personnel to work with these materials. The safety devices include the building structure, shielded hot-cell enclosures, ventilation and air-cleaning equipment for the hot cells and the building areas, radiation-detection instruments, remote manipulating devices, glove boxes, shielded viewing systems, hot waste systems, and fire-protection systems.

Hot-laboratory operations involve hazards that are sometimes of great magnitude. The effects of an "incident" or accident could be fatal, costly, and widespread since most of the hot-laboratory buildings also house offices, "cold" laboratories (for direct handling of nonradioactive and nontoxic materials), and other uncontaminated areas. Although such uncontaminated areas are protected by "clean" (uncontaminated) corridors, by barrier walls that have few penetrations, and by differential-pressure atmospheres, they could become contaminated by an accident. These clean facilities, which are required in the technical support of the hot laboratories, whether located in the same building, in attached buildings, or even in adjacent buildings, add many more people who may be exposed in the event of an accident. In addition, they add space and equipment that would be costly to decontaminate. The time lost from an accident would also add to the ultimate expense of the incident.

Most of the literature on hot-laboratory design and the attendant safety considerations is concerned with laboratory complexes that include hot, cold, and supporting facilities in the same building. The proceedings of the Hot Laboratories and Equipment Conferences[22–29] contain many papers that describe designs of such facilities. This review is concerned with three recent discussions of safety in hot laboratories: (1) the safety aspects of the BNL incident of May 1957, which involved corrosive fumes; (2) procedures for use in the handling of plutonium; and (3) the safety aspects of the ORNL incident of April 1960.

BNL Incident of May 1957

The so-called "BNL Incident of May 1957" was described by Stang at the Sixth Hot Laboratories and Equipment Conference,[27] and the safety recommendations that resulted from the study of this event were discussed. The accident took place in a nonradioactive pilot-plant arrangement that was located in a hot area of the laboratory but was not contained in a hot cell. This pilot plant was being used for tests of a volatility process of recovering uranium, in which bromine trifluoride was employed to fluorinate the metal. The incident apparently resulted from a piece of uranium metal being accidentally withdrawn into the gaseous region above the liquid in the dissolver. A series of about five explosions occurred in rapid sequence. Highly corrosive fumes of the bromine trifluoride then penetrated many parts of the building.

Detailed examination of the facts surrounding this incident and of the subsequent events led to a number of recommendations with regard to future operation. Perhaps the most significant conclusion was that a hot-laboratory building should not be employed for purposes other than those for which it was designed. Pilot-plant work of this nature should be isolated in a separate, sealed building. It was specified that the BNL hot-laboratory facility was not again to house a pilot-plant operation involving an explosion hazard, regardless of the number of safety precautions taken to prevent explosion.

The characteristic behavior pattern of people immediately following an incident was well illustrated in this case. The first reaction of an individual is to run away from the scene and then to take a shower. In the BNL incident, personnel followed this pattern, and in their excitement some even ran past one or more safety showers in an adjacent safe area to a shower which was familiar to them. This behavior pointed up several considerations in the design and placement of safety showers. Stang's[27] suggestions are summarized below:

1. Safety showers should be located in all areas of the hot-laboratory building.

2. Some safety showers should be located outside the hot-laboratory building since a fire or explosion can quickly fill an enclosed area

with blinding or toxic smoke or fumes and thus prevent the use of indoor showers.

3. Safety showers should be marked with bright-green flashing lights.

4. These lights, together with other desirable emergency lighting, should be on an emergency power circuit so that they will remain on at all times, regardless of power failure or intentional shutoff of normal power.

5. Eyewash fountains should be installed at all safety showers.

6. Everyone in the building should participate in a regularly scheduled safety-shower check. Scientists, technicians, supervisors, secretaries, janitors, and others should take turns in testing all the showers at scheduled times.

In addition, it is the opinion of the reviewer that the shower valve-actuating mechanism should be a wall-fastened chain so that persons who are unable to see can grope along the wall to find it.

In the BNL incident, a partially opened door equipped with a door closer in poor working condition contributed to the spread of fumes and smoke. Doors and door closers should be in good operable condition. Door closers are frequently installed to position doors in a manner that will ensure proper flow of ventilating air from cleaner areas. The effects of any imbalance due to explosions or sudden drafts will thus be minimized by properly operating doors. Furthermore, all doors, both exterior and interior, should be readily operable from both sides. If normal traffic is to be limited to one direction, a key (under glass) or other device at the door frame should be available for emergency entry.

In the BNL incident, no radioactivity was involved. However, the fumes were highly corrosive. Rapid purging of the smoke was necessary to clear the air in the building in order to search and guarantee that no one was left in the building and also to permit the fire fighters to extinguish the small fire. It was difficult to get to the two override manual switches for the ventilation fans, which were located in the smoke-filled area. Stang[27] suggested that a manual override switch for the building ventilation system should be located on the outside of the building. Override switches for air-conditioning systems, which can very effectively scatter radioactive particles throughout a building, and for other critical services should also be installed on the outer walls of the hot-laboratory building. It is now considered good practice, in order to confine contamination, to clean all cell ventilation air before it leaves the laboratory building and enters the plant-wide system.

In order to ensure that there is always, during working hours, one person who is in charge of the hot-laboratory facility, a check-board system was devised at BNL. Six persons are available on the permanent emergency staff. When the person at the top of the list leaves the building, his departure is noted on the board, the next person is notified that he is the emergency supervisor. The responsibility is passed along, starting with the first name each morning.

A rule has been set up to require everyone, including transient visitors, to wear safety glasses at all times in the hot-laboratory building, except in the offices and in the cold corridor. This replaces a previous rule requiring safety glasses to be worn only by those performing hazardous operations and those in the vicinity of such operations. It was found that this rule had left too much to the judgment and discretion of individuals. One-piece plastic suits that cover the entire body and are equipped with a bottle containing a 10-min supply of compressed air are available in strategic areas and are ready to step into at a moment's notice.

In addition to these major recommendations, a number of other safety suggestions that are applicable to all hot laboratories were made by Stang.[27] His suggestions are summarized below:

1. A public-address system is useful on many occasions, and, in an emergency, it becomes very valuable; the system should be set up to operate from an emergency power supply.

2. Scheduled reindoctrination of hot-laboratory personnel in the use of emergency equipment should be practiced since people tend to forget over a period of time how to use some of the devices properly.

3. Exterior electric outlets for operating portable pumps for emergency air sampling and monitoring should be available.

4. It should be possible to hear fire alarms in all areas of a facility.

5. Since the accounting for personnel after an emergency can be no better than the knowledge of whom to expect to have been in the

building, procedures should be established for assuring accurate knowledge of personnel occupancy of the building at all times. (At BNL this problem was difficult to handle because about 40 per cent of the occupants of the building were organizationally outside the hot-laboratory division. A picture chart is now hanging in the staff office of the building, and both the temporary personnel using the facilities and the permanent employees are shown on this chart. Absentees and temporary absences from the building are posted. Frequent drills have demonstrated that, with the use of these aids, all occupants of the building can be readily accounted for.)

6. All new temporary and permanent occupants of the building should be issued safety instructions, building rules, safety shoes, and safety glasses.

7. In order to make safety posters and other important signs stand out emphatically, all unnecessary or unimportant signs should be removed.

Dry Boxes for Use with Plutonium

Safety considerations involved in the handling of plutonium in a dry-box system were also discussed at the Sixth Hot Laboratories and Equipment Conference.[27] It was pointed out that the principal hazards in the handling of plutonium and its alloys are inhalation and fire; shielding from radiation is usually not a problem, although it may become one when handling plutonium of high exposure from spent power-reactor fuels. In order to protect personnel from the inhalation hazard, the handling is done in an airtight glove box or dry box. The interior atmosphere of the box is usually maintained at a negative pressure of approximately ½-in. H_2O so that leakage, if any, is inward.

Because the consequences of the spread of contamination from a fire would be serious, all possible precautions are taken in the design and construction of such handling facilities to prevent fires. Fire-resistant construction, metal equipment, fire-detection and alarm systems, fire walls and doors, and the practice of emergency procedures are recommended. In order to minimize the fire hazard, dry inert atmospheres containing a minimum of 30 per cent helium are maintained in the boxes. A helium analyzer is used in the exhaust to provide continuous monitoring to assure that this condition is met. The presence of combustible vapors in dry boxes is prohibited because such vapors have caused explosions in the past. Sprinkler systems are excluded from plutonium-handling areas.

The recommended dry-box design provides interiors that are smooth and free of crevices, with rounded corners to minimize collection of contamination and to facilitate cleaning. No sparking electrical equipment is permitted in the interior of a box. The box is illuminated from the outside through windows. For the windows of dry boxes, polymerized polyester resin (Homolite) is recommended since it is believed to be the most resistant of available materials to pressure and to chemicals. Where the work being done requires visual quality control (such as machining and precision measurements), laminated safety glass is used.

Individuals involved in the handling of plutonium should be made fully aware of the hazards. They should undergo regular health checks, and they should receive safety instructions in operating and fire-handling procedures.

(Frank Ring, Jr.)

ORNL Incident of April 1960

The events leading to a release of activity from a hot cell at the Oak Ridge National Laboratory, the subsequent handling of the incident, and the recommendations of the investigating committee are given in an as yet unpublished report.[30] The accident occurred when the pressure in the cell was permitted to exceed that of the operating face area. A total of less than 1 curie of beta-gamma fission-product activity from a sectioned fuel element was expelled into an operating area. From this area the activity was sucked down an adjacent stairwell through an open doorway in a wall separating the office section of the building from the hot area. A large fraction of the activity was picked up by the air conditioners at the head and foot of the stairwell, and, although the air was filtered, some of the activity was circulated back over the hot-cell area.

At the time of the activity release, there were eight persons in front of the cell and directly in the path of the radioactivity. Another 13 persons were in the general hot-cell area but not directly in the path of the activity. None of the exposed individuals is believed to have received in excess of the maximum permissible

exposure. Analyses of the presently available exposure data, which must still be considered to be preliminary, indicate that one person ingested a quantity of activity which will give a 1.2-rem exposure to both lungs and bone during the first quarter and 2.1 rem during the first year. Another person will receive a 0.3-rem exposure during the first quarter, and the other individuals will receive at least a factor of 10 less than the highest exposure.

Although the cause of the positive pressure in the hot cell is not known with certainty, it is now believed to have been the result of the following sequence of events: (1) openings in the face of the hot cell in question were not sealed, (2) alterations were being made in an adjoining cell and that cell was open on one side, (3) the two cells were interconnected, (4) an outside door opposite the open side of the cell being altered was opened, and (5) pressure disturbances that were created by the opening of the door and were transmitted through the two cells caused activity to blow out through the unsealed penetrations in the operating cell face.

It is apparent that (1) the operating cell was not adequately equipped to maintain the required negative pressure under the conditions to which it was subjected and (2) there was no alarm system to indicate the failure of the protective pressure differential. Furthermore, it was imprudent to permit alterations to be carried out in an adjoining cell without sealing the openings (about 3 sq ft in total area) between the two cells. The following statements are excerpts from the conclusions and recommendations of the investigating committee report:[30]

1. Prior to the incident, no one properly assessed the extent of this hazard. ... [The] procedures and equipment ... were inadequate to cope with the unusual hazard involved.

2. The facility, in addition to not being equipped for such an operation, was in a dangerous state. For instance, the off-gas filters were bad, the adjoining cell was wide open, the cell pressure was dangerously near that of the operating area, and the back door to the building was left open.

3. The flow and recirculation of air throughout the entire building, from hot-cell areas to office-laboratory areas, was contrary to good practice; and previous incidents somewhat similar to this one, but of lesser extent, were common knowledge of persons familiar with this area.

4. Inadequate radiation-monitoring equipment was available to determine the extent and nature of the activity released.

5. No emergency communication system was available, and at least 15 min was required to alert the building occupants that a radiation release had occurred.

6. No masks or other emergency supplies were readily available; and, after they were aware that air-borne activity was present, five persons remained in the area without masks [in order] to secure the cells.

As a consequence of this incident, a set of criteria for the safe operation of hot cells has been established. These criteria are based on principles that have been recognized for years, but they are specific with respect to the application of these principles. In addition to establishing these criteria, a system has been specified for ensuring compliance with the criteria for every new and different situation, for establishing adequate emergency procedures, and for training an emergency organization.

The outline of minimum standards for hot cells given in the ORNL investigating committee report is listed below:

1. A minimum of 0.1 in. of water vacuum must be maintained on hot cells and a minimum flow through any opening of at least 100 ft/min with sufficient exhaust capacity so that these minimum values can be maintained for any credible situation that may arise, and an alarm must be provided to indicate when the vacuum drops below 0.1 in. of water.

2. Exhaust from hot cells must pass through at least one roughing filter and an absolute filter and be continuously monitored with an alarm before being discharged to the environment.

3. For any operation which is likely to produce gross amounts of air-borne contamination, an additional line of containment within the cells should be provided with an exhaust which is filtered before discharging into the cell ventilation or other suitably filtered system.

4. Provision must be made for removing chemical contamination from the cell off-gas such that the filters are not impaired, and special chemical-resistant filters are to be used where needed.

5. Vacuum lines run into cells should be adequately trapped, treated for chemical contamination, and filtered before leaving the shielded area.

6. All known liquid radioactive effluents should be routed to the appropriate hot waste system.

7. All other possible routes of radioactive material release, such as process water drains within the cells, should be monitored, and provision should be made for diversion to the hot waste system. Process water should not be discharged to the hot waste system except in case of an emergency.

8. All operations in hot cells which involve unusual hazards (such as large quantities of alpha or soft-beta activity which are difficult to detect; gaseous activity; possibility of an uncontrolled chemical reaction, fire, or explosion; and possibility of a critical assembly) should be individually reviewed by the director of Radiation Safety and Control.

9. The area around hot cells up to a second line of contamination control should be regarded as a likely contamination zone, and monitoring for all persons and materials leaving this clearly defined zone should be mandatory.

10. Adequate radiation-detection instruments should be present in such areas to give prompt indication of air-borne activity, any significant spread of contamination, or an increase in background activity.

11. In addition to a line of contamination control, provision for a complete second line of containment should be provided around hot-cell areas, such that, when a radiation release is detected, the area is automatically isolated and exhausted through a filter, if practical the cell ventilation system.

Conclusions

It is apparent that dry boxes and hot cells can be constructed and operated safely. The incidents discussed here and the supporting literature show the way to attain safety in a hot-laboratory complex. If accidents in such facilities are to be held to a minimum, it is imperative that the principles illustrated here and reiterated in the recommendations be applied in the design and operation of such facilities.

For many years it has been the practice in industry to separate, confine, and usually contain potentially hazardous operations. As a corollary, the hot-laboratory building should contain *only* hot cells or glove boxes, ventilating filters and fans, air-conditioning equipment, and a radioactive waste system. The supporting facilities should be in a *separate* building. This philosophy of design should be seriously considered for any new hot-laboratory facilities, especially in view of the increasing levels of activity being handled in hot cells, and the recent incidents.

Containment facilities within the hot laboratory should be designed to provide two lines of defense, each "failure free" and each provided with instrumentation and alarms that would direct attention to an incipient failure in any of the containment provisions before the complete failure of either line of defense. Such provisions would include the maintenance of an adequate negative pressure differential from the outside to the inside of a hot-laboratory area, as well as from the laboratory area to the hot cell. These pressure differentials should be attained with facilities designed so that the failure of any single blower or power supply would not result in the loss of the desired pressure differentials, which, in turn, should be monitored in such manner as to instigate an alarm at a significant loss-of-pressure differential.

Although such provisions should be required for high-level (≫1 curie) hot cells, it would not be correct to assume that these provisions would be required for all hot cells regardless of the nature and amount of the activity to be handled. Rather, the consequences of the accidental release of activity from a given facility and for a particular operation must be evaluated. The adequacy of any facility for any particular operation must be determined in view of the consequences of the credible activity release. There is considerable judgment involved on the part of the facility operator, both in the assignment of a maximum credible accident and in the determination of acceptable consequences for particular facilities. Thus it may be shown that there will exist some low activity level below which a single containment barrier (in this case, the pressure differential maintained between the laboratory area and the hot cell) will be sufficient.

Once the safety criteria for a facility have been determined, it is the responsibility of the operator to see that the safety features are operative, adequately backed up, and instrumented. In addition, the operator must establish procedures to familiarize the operating personnel with the operational hazards and emergency actions. (W. B. Cottrell)

Containment of Power Reactors

Every existing and presently proposed U. S. power reactor has a separate containment structure that is functionally independent of the reactor and acts as a final barrier to the spread of hazardous radioactive contaminants to the surroundings in the event of accidental release of radioactivity from the enclosed reactor system. In most cases this containment structure has taken the form of a steel pressure vessel that is capable of withstanding the maximum

pressure anticipated during a maximum credible accident.

There have been a few deviations from this concept in attempts to reduce costs. For instance, the Boiling Nuclear Superheater Reactor (BONUS) to be built at Punta Higüero, Puerto Rico, will have a large-volume steel hemispherical dome welded to a cylindrical section resting on a thick concrete foundation mat. In this case the philosophy is to keep the pressure low by enclosing the whole plant in a large-volume structure, and, at the same time, to reduce the number of penetrations of the containment shell. For the boiling-water-reactor plant at Humboldt Bay, it is proposed to employ a pressure-suppression scheme in which the steam from fissure of the main coolant system will be routed through a water tank and condensed.

The Hallam sodium-cooled reactor will be contained by individual steel-lined concrete cells within which the primary system components are located and which are maintained with a nitrogen atmosphere. Additional containment is provided by a conventional metal-covered steel-frame building which is equipped with a blower exhaust system capable of maintaining a slight negative pressure within the building. Exhaust from the blowers is filtered before discharge to the atmosphere through a stack. This type of containment was discussed more fully in a previous issue of *Nuclear Safety*.[31]

In contrast to these attempts to reduce costs by reducing pressures, the SPWR and the CVTR will be housed in containment buildings which, like steel pressure vessels, are designed to withstand full pressure in the event of an accident but which will be less costly because of the use of different methods of construction. The proposed containment shell for the SPWR consists of a steel shell faced on both sides with reinforced concrete. The structure is designed so that the load is evenly divided; that is, one-half is to be taken by the steel shell, and the other half is to be taken by the reinforced-concrete structure.

The containment building for the CVTR will be fabricated of reinforced concrete lined with $\frac{1}{8}$-in. sheet steel in the cylindrical sections, and it will be capped with a hemispherical steel head. None of the presently proposed plants use prestressed concrete, although it will be used in the Heavy Water Components Test Reactor at Savannah River. The use of concrete for containment buildings and pressure vessels was discussed previously in *Nuclear Safety*.[32]

In some cases, factors exist which make it advantageous to maintain a reasonably high-pressure atmosphere around the reactor. For example, the FWCNP, being CO_2-cooled, is enclosed in a standard cylindrical steel pressure vessel. In contrast to the large-volume building designed to decrease the pressure of BONUS, the containment shell for the FWCNP is designed with a small volume to assure a reasonably high pressure of 27 psig after an accident so that the main coolant blowers can continue to supply sufficient coolant to remove afterheat from the core. Of course, this incentive exists only for gas-cooled reactors.

In addition to variations in the type of containment structure as dictated by pressure criteria, there are many variations in leakage-rate tolerances, in methods for sealing penetrations, and in provisions for shock-wave and missile protection. The information available in the hazards reports for the currently operating and proposed U. S. power reactors has been summarized in Table IV-1 in order to point up these variations. (The data in Table IV-1 were obtained from the hazards reports, most of which are listed in Section III, references 14 to 29.) As indicated, most pressurized-water and boiling-water plants are contained in steel pressure vessels, and most of the containment structures are designed to withstand the pressure generated by either a release of the primary coolant or the primary coolant plus a fraction of the secondary coolant while expanding adiabatically to equilibrium with the atmosphere within the containment structure. In order to be conservative, credit is not usually taken for the effects of pressure-reduction systems or the cooling effects of the shell and the enclosed equipment. The systems that include means for pressure reduction are indicated in Table IV-1.

The quite different containment structures designed for the two sodium-cooled reactors, Hallam and Enrico Fermi, emphasize the effects of the differences in the postulated maximum credible accidents. The sodium coolant in these reactors is circulated at near atmospheric pressure, but, as is well known, sodium is extremely reactive with air. By operating

Table IV-1 CONTAINMENT STRUCTURES OF VARIOUS NUCLEAR POWER PLANTS*

Table IV-1 CONTAINMENT STRUCTURES OF VARIOUS NUCLEAR POWER PLANTS*

Short name of plant	Type of containment vessel and dimensions
PWR	Four steel-shell compartments: two horizontal-cylinder boiler chambers, 1.25-in.-thick walls, 50 ft in diameter, and 97 ft long; one auxiliary chamber, 1.25-in.-thick walls, 50 ft in diameter and 147 ft long; one reactor chamber 0.75–1.00-in.-thick walls, 38 ft in diameter with an 18-ft-diameter by 20-ft-high cylindrical dome on top
Indian Point	Spherical steel shell, 160 ft in diameter; upper portion 0.89 in. thick, lower portion 1.03 in. thick
Yankee	Spherical steel shell, 125 ft in diameter; [minimum plate thickness is $\frac{7}{8}$ in., penetrations reinforced to full strength; maximum thickness is 1.25 in.]
N.S. *Savannah*	Horizontal steel cylinder (axis fore and aft) with hemispherical heads; cylinder 35 ft in diameter and [50.5] ft long; walls of 1.25 in. minimum thickness; cupola provided to house control-rod drives; cupola 13.5 ft in diameter and 16.5 ft high
Saxton	Steel cylinder 56 ft in diameter and 101 ft high
SPWR	Steel cylinder, 53 ft in diameter and 115.75 ft high, with hemispherical head and torispherical bottom; steel walls 0.353 in. thick; steel encased in reinforced concrete, 1.5 in. thick on inner surface, 2 ft thick on outer cylindrical surface, 1.5 ft thick on outer surface of hemispherical top, and 3 ft thick on outer surface of torispherical bottom
VBWR	Steel cylinder, 48 ft in diameter and 110 ft high, with hemispherical heads; sides 0.875 in. thick; top and bottom 0.436 in. thick
Dresden	Spherical steel shell, 190 ft in diameter; upper portion 1.25 in. thick; lower portion 1.40 in. thick
Elk River	Steel cylinder, 74 ft in diameter and 115 ft high, with hemispherical top and ellipsoidal bottom; sides 0.875 in. thick; top 0.5 in. thick; bottom 0.70 in. thick
Pathfinder	Steel cylinder, 50 ft in diameter and 120 ft high, with hemispherical top and ellipsoidal bottom; sides and bottom [$1\frac{3}{8}$] in. thick; top [$1\frac{1}{16}$] in. thick
Humboldt Bay	Steel-lined concrete (4 ft thick) pressure-suppression chamber; dry-well reactor chamber, 17 ft in diameter and 67 ft high
BONUS	[Steel hemispherical dome, 166 ft 8 in. in diameter with 24-ft-high steel cylindrical side wall erected on a 182-ft-diameter concrete foundation mat; steel containment structure sealed to concrete floor; top of dome 107 ft 4 in. above floor; all plant components enclosed by dome; dome steel $\frac{5}{16}$ in. thick, cylindrical steel wall $\frac{13}{32}$ in. thick; 24-ft-high concrete wall around outside perimeter of steel cylindrical wall retains earth embankment for hold-down of containment building]
Big Rock Point	Spherical steel shell 130 ft in diameter
Hallam	Reactor enclosed in steel-lined cylinder, 22.5 ft in diameter and 36 ft high, covered with 6-ft-thick shield plug; steel liner 0.5 in. thick; conventional steel-sandwich building with negative pressure maintained by ventilation through particulate filters and up stack
Enrico Fermi	Steel cylinder, 72 ft in diameter and 120 ft high, with hemispherical top and ellipsoidal bottom; sides 1.03 in. thick; top 0.52 in. thick; bottom 1.03 in. thick (1.25 in. at knuckle line)
EGCR	Steel cylinder, 114 ft in diameter and 189 ft 7 in. high, plus separate steel-lined experimental equipment cells for in-pile loops
FWCNP	Steel cylinder, 87 ft in diameter and 118.5 ft high, with hemispherical top and dished bottom
Peach Bottom	Steel cylinder (dimensions not available)
Piqua	Steel cylinder, 73 ft in diameter and 126.5 ft (from drawing) or 112 ft (from text) high, with hemispherical top and dished bottom
CVTR	Reinforced concrete cylinder lined with $\frac{1}{8}$-in.-thick steel, 58 ft in inside diameter and 146 ft high (inside); hemispherical head lined with $\frac{1}{2}$-in.-thick steel; flat bottom

*Entries in brackets represent changes made by the reactor designer since the publication of the hazards reports used for this comparison. Since the reviewer has no additional information explaining such changes, these changes are only listed for additional information.

Table IV-1 (Continued)

Short name of plant	Distance from grade elevation to bottom of structure, ft	Structural material	Applicability of ASME Code	Volume of containment structure, cu ft	
				Gross	Free
PWR	−52	A201-A300 steel	Yes	635,000	473,000
Indian Point	−75.5	A201-A300 steel	Yes	2,140,000	Not available
Yankee	+24	A201-A300 steel	Yes	1,020,000	840,000
N.S. *Savannah*	Not applicable	[A212-B]	Yes	Not available	32,300
Saxton	Not available	Not available	Yes	Not available	Not available
SPWR	Not available	Steel and concrete	No	Not available	Not available
VBWR	−35	SA212 steel	Yes	160,000	Not available
Dresden	−39	A201-A300 steel	Yes	3,600,000	2,880,000
Elk River	−18.5	A201-A300 (57TGB) steel	Yes	405,200	287,000
Pathfinder	−65	SA212-B-SA300 steel	Yes	[212,000]	142,000
Humboldt Bay	−67 (approx.)		[Dry-well suppression, yes; suppression chamber, no]	Not available	[12,500](dry well) [33,400 suppression chamber]
BONUS	−9	Dome stiffeners: ASTM-A131 steel Dome plate: ASTM-A201 steel	No, API tentative standard No. 620 and ASA N6	1,780,000	Not available
Big Rock Point	−27	SA201-B-SA300 steel	Yes	1,153,000	922,000
Hallam*	[−47]	Liner: A212 steel		1,870,000 (main and auxiliary bays); 15,000 (gas volume in cells)	Not available
Enrico Fermi	−51	A201-A300 steel	Yes	415,800	
EGCR	−73 (approx.)	SA212-B steel	Yes	Not available	1,360,000
FWCNP	−15 (approx.)	A201-B-A300 steel	Yes	Not available	Not available
Peach Bottom	Not available	A201-B fine grain	[Not decided]	Not available	Not available
Piqua	−55 (from drawing) −40 (from text)	Not available	Yes, not stamped, design pressure <15 psig	Not available	Not available
CVTR†	−47 [−54](drawings)	Liner: A285-B [A285-C] sheet steel	No [Yes]	376,000 [281,000]	320,000 [234,000]

* The definition of containment is somewhat arbitrary here, depending upon whether the building or the cells are concerned.

† The changes in values for the CVTR are a result of modification in container design.

Table IV-1 (Continued)

Short name of plant	Properties of reactor coolant				Energy released to containment structure in event of an accident, Btu		
	Type	Quantity, lb	Operating pressure, psia	Operating temperature, °F	Stored energy ($\times 10^6$)	Chemical reaction energy ($\times 10^6$)	Nuclear and other energy from accident ($\times 10^6$)
PWR	H_2O	[131,000]	[1800]	[500]	78	None	4.6
Indian Point	H_2O	163,000	1500	500	130.5	None	4.3
Yankee	H_2O	186,000	2150 (max.)	518	92.2	None	None
N.S. *Savannah*	H_2O	77,000	1750	508	[37.9]	[0.05]	[9.2]*
Saxton	H_2O		2000	503	Not available	Not available	Not available
SPWR	H_2O		1700–1900	520	Not available	Not available	Not available
VBWR	H_2O	35,000	1000	545	20	None	None
Dresden	H_2O	376,000	1000	545	200	None	None
Elk River	H_2O	27,957	936.5	540	13.6	None	None
Pathfinder	H_2O	100,000	600 (in reactor) 540 (at outlet)	489 (boiling region) 825 (superheater)	47.4	None	None
Humboldt Bay	H_2O	[82,000]	[1025]	[548]	36	None	[5]
BONUS	H_2O		915	534 (boiling region) 900 (superheater)	15.3	2.4 (taken by moat)	Released by natural convection from building
Big Rock Point	H_2O		1050–1500	344.7–550	Not available	None	None
Hallam	Na		Not pressurized	~778 (primary) [945] ~727 (secondary) [895]	None	None	None
Enrico Fermi	Na		Not pressurized	[800 (primary)] [750 (secondary)]	Sodium-air reaction, nuclear heating, and afterheat being considered		
EGCR	He	1,350	315	780	Not available	None	None
FWCNP	D_2O CO_2	90,000 15,000	50 500	231 (moderator) 800 (reactor)	~10 (in CO_2) + 4 (in steam and heat-exchange coils)	None	None
Peach Bottom	He		[350]	[660 inlet, 1380 outlet]	Not available	Not available	Not available
Piqua	Santowax R	76,450	120	549	Not available	Not available	Not available
CVTR	D_2O	15,370	1500	530	7.45	2.97	1.00

*Containment-vessel pressure would be 220 psig when this value is added.

Table IV-1 (Continued)

Short name of plant	Containment-structure design pressures, psig			Leakage rate at design pressure, %/day	
	Positive	Negative	Strength test (pneumatic)	Specified	Observed
PWR	52.8	Not available	70	0.1	0.15 ± 0.075
Indian Point	27.5 (T = 225°F)	Not available	34.4	0.1	
Yankee	34.5 (18 sec after rupture, T = 249°F)	Not available	40	0.1	
N.S. *Savannah*	173 [186]	Not available	186 [173] (hydrostatic)	0.2	
Saxton	28.3	Not available	35	0.2	
SPWR	30 (T = 248°F)	Not available	37.5	0.5	
VBWR	45	Not available	56.25	1.0	<0.02
Dresden	29.5 (T = 325°F)	1 (vacuum-break valves set at 1 psi, annunciator at 0.75 psi)	37	0.5	0.016
Elk River	21	0.5 (vacuum-break valves set at 0.33 psi)	26.25	0.1	
Pathfinder	78	[Vacuum-break]	97.5	[0.2]	
Humboldt Bay	40–67 (dry well) [72] 10 (suppression chamber)	Not available	1.25 × design pressure	0.5 (suppression chamber) [0.05 dry-well vessel]	
BONUS	5 (T = 140°F)	0.25 (vacuum-break valves installed)	6.25	0.2	
Big Rock Point	27	Vacuum-break valves installed	33.7	0.5	
Hallam	None [2 psig]	$-\frac{1}{8}$ to $\frac{1}{2}$ (H_2O pressure)	[$2\frac{1}{2}$ psig (reactor cavity)]	320 (building) [0.001 (reactor cavity)]	
Enrico Fermi	32	2	40	0.06	
EGCR	12	Vacuum-break valves installed	15	~2 (based on rupture of one contaminated in-pile loop)	
FWCNP	27 (T = 450°F)	0.25 (vacuum-break valves installed)	34	0.25	
Peach Bottom	[About 5 psig]	0.2 (vacuum-break valves installed)	1.25 × design pressure	[0.2]	
Piqua	5 (T = 125°F)	Not available	6.25	0.2% per day per psig (max.: 5% per day at 5 psig)	
CVTR*	14 [21] (T = 120°F) [216.5°F]	Not available	17.5 [26.25]	0.1	

* The changes in values for the CVTR are a result of modification in container design.

Table IV-1 (Continued)

Short name of plant	Leakage tests used	Penetrations of containment structure*
PWR	Reference pressure drop, soap solution, Freon or helium injection	Ventilation ducts, normally open, six personnel air locks, pipe penetrations
Indian Point	Soap solution	Ventilation ducts, normally closed; no personnel allowed during operation; main steam lines close automatically on rise-of-pressure signal in shell; smaller lines actuated manually
Yankee	Continuous reference pressure drop, soap solution, Freon injection (optional)	48 pipe penetrations; openings normally closed during operation; process lines have two check valves in series; closure valves operate at 5-psi ΔP; electrical penetrations leak tested at −40 to − 140°F
N.S. *Savannah*	Not available	Two 42-in.-ID access hatches and two manholes; [cupola 13.5 ft, cupola hatch, 5.6-ft inside diameter, ventilation inlet and exhaust] shell will be sealed during operation
Saxton	Not available	All openings shut during operation, except process lines, and no personnel allowed during operation; no personnel or equipment airlocks, bolts on doors are used instead; two check valves in process piping
SPWR	Soap solution before pouring concrete, pressure drop afterward	Fresh air continuously passed through containment shell; in case of accident, air will be shut off by automatic butterfly and backup valves
VBWR	Soap solution, reference pressure drop	Since both doors of access lock must be open when transferring fuel casks, no operation, refueling, or other work allowed during this time; large removable pieces require unbolting of hatch; no containment necessary with clean-core critical experiments
Dresden	Soap solution, reference pressure drop	[Ventilation ducts and other critical penetrations close automatically upon actuation of the high sphere pressure or low primary drum system, or by manual actuation]
Elk River	Soap solution, reference pressure drop	Personnel airlock, equipment airlock, and freight doors
Pathfinder	Soap solution and tracer gas if reference pressure drop test shows >0.2% leakage [0.27%]	Access allowed during operation; normal and emergency airlocks provided; emergency airlock inner door normally open; once-through ventilation system shut by butterfly valves with rubber seats; operating lines have isolation valves

* It is impractical to attempt to list the numerous details here.

Table IV-1 (Continued)

Short name of plant	Leakage tests used	Penetrations of containment structure
Humboldt Bay	Reference pressure drop; soap solution or halogen injection	Ventilation air routed from personnel access points to unoccupied areas subject to potential contamination and then up stack; contaminated air will be exhausted through scrubbers (5% aqueous NaOH)
BONUS	Vacuum box, soap solution and integrated leak rate; vacuum box test on cylindrical wall prior to construction of exterior retaining wall	Nine electrical and 30 mechanical penetrations, including ventilation air intake and discharge ducts; two personnel airlocks and one freight door; ventilation ducts normally open; ducts close automatically in case of accident
Big Rock Point	Not discussed	30 power, control, and instrument cables; two valves (one automatic and one manual) on all lines to radioactive sources, each on opposite side of containment wall; ventilation normally once-through
Hallam	Cavity liners, helium injection, and soap solution	Conduit to primary Na cells; once-through ventilation with filters in main building*
Enrico Fermi	Soap solution, reference pressure drop	500 penetrations; seals designed to withstand 460°F wall temperature and 1240°F gas temperature; ventilated above operating floors
EGCR	Not available	76 penetrations; automatic valves or check valves will be provided; ventilation system will be closed by quick-closing butterfly valves
FWCNP	Soap solution	Personnel will be admitted during operation; two quick-closing, automatic valves in series in process lines
Peach Bottom	Not available	Ventilation ducts [service water and air, radioactive waste, instrumentation, lube oil, trap services] and steam lines; personnel allowed during operation
Piqua	Soap solution, reference pressure drop	Ventilation ducts, normally open; isolation valves provided in case of accident when inleakage through building is routed through prefilter and high-efficiency filter to stack; total number of penetrations not known; however, there are at least 78 conduits and 66 pipes
CVTR	Freon injection, soap solution	All penetrations closed, and no one allowed in during operation

* See the first footnote on page 58.

Table IV-1 (Continued)

Short name of plant	Property sensed by system for sealing penetrations	Missile protection	Shock-wave protection
PWR	Not available	Ductile materials used throughout to minimize formation of missiles; no other provisions	Not discussed
Indian Point	Pressure increase	No provisions; probability of credible missiles from within too small	Not discussed
Yankee	Pressure increase	Ductile materials of construction; reinforced concrete around reactor; 1- to 2-in. steel plate above control-rod drive mechanism plus 18 in. of concrete	Not discussed, but reinforced concrete around core is designed to 6 psi
N.S. *Savannah*	[Pressure increase to close pipe penetrations]	Shell will hold 20-lb 2-in.-diameter thermometer-weld missile; ductile material used throughout system	Not discussed
Saxton	Not available	Greater than 3-ft-thick biological shield	Not discussed
SPWR	Pressure increase	Ductile materials; 18-in. concrete liner where uncovered; 3 ft to operating deck, 2 ft to dome, 1.5 ft over dome	Not discussed, but inherent in large amounts of concrete on each side of steel
VBWR	Not available	4-ft-thick reinforced-concrete biological shield; when power is greater than 10 Mw, 18-in.-thick steel missile shield in place above reactor	Not discussed
Dresden	Not available	Not available	Not discussed
Elk River	Activity increase	Inner containment wall lined with concrete; reactor enclosed in 5 to 8 ft of concrete; top 114,000-lb shield plug will not become missile because of insufficient rate of steam release	Not discussed
Pathfinder	[Activity release only]	Shield pool above core; concrete around sides; high-speed rotating machine not inside building	Not discussed
Humboldt Bay	[Dry-well high pressure, reactor low water level, main steam-line break]	[Dry-well steel liner backed with concrete]	[Shock pressure for MCDA 271 psia*]

[* The maximum calculated shock-wave pressure is less than the calculated rupture jet impact pressure of 1640 psi.]

Table IV-1 (Continued)

Short name of plant	Property sensed by system for sealing penetrations	Missile protection	Shock-wave protection
BONUS	Pressure and activity increases	Missile shield around turbine; otherwise not discussed	Not discussed
Big Rock Point	Pressure increases, low water level, power loss	Concrete shield around reactor	Not discussed
Hallam	Not available	No special provisions	Not discussed
Enrico Fermi	Not available	Concrete around core	Effects of 300- and 1000-lb TNT equivalent were analyzed; no permanent yield expected at 300-lb TNT; no rupture at 1000-lb TNT
EGCR	Not available	Missiles will be contained by reinforced-concrete biological shield, except in service-machine room, steam generator room, and blower room; protection will be provided if further analysis shows it to be necessary	Not discussed
FWCNP	Not available	Will be studied	Not discussed
Peach Bottom	[Pressure and activity release]	[Will be studied]	Not discussed
Piqua	Activity increase	8-in.-thick concrete shield inside shell, 6 in. thick at top	Explosive hazard of Santowax-R dust exists at 35,500 mg/m^3; dust concentrate kept at 94 mg/m^3, well below this value
CVTR	Not available	Primary biological shield, 2½-ft concrete operating floor, and 2-ft concrete housing above steam generator; study being continued	

Table IV-1 (Continued)

Table IV-1 (Continued)

Short name of plant	Pressure-reduction method	Weld inspection techniques	Provisions for temperature control of structure
PWR	None	Welds radiographed and/or magnetic-particle tested; [dye penetrant test]	Concrete insulation outside; internal temperature between 50 and 122°F [air-cooling system]
Indian Point	Water spray provided, but not considered in sizing containment vessel	Weld X-rayed and pneumatic tested	Not discussed
Yankee	None		Atmospheric control inside (50 to 125°F)
N.S. *Savannah*	None	All welds X-rayed	[Containment cooling system and outside ventilation]
Saxton	Spray provided, but not considered in design	Not available	Atmospheric control inside (50 to 122°F)
SPWR	None at this time	Soap solution and radiographic tests before concrete is placed	Atmospheric control; 60 to 120°F inside when outside of concrete is at ~0°F
VBWR	Considered, but not used		
Dresden	Cooling spray, giving 30×10^6 Btu/hr, provided	Radiograph	Thermal insulation outside; internal heat keeps shell at 30°F when −20°F outdoors with 15-mph wind
Elk River	15,000 gal of borated water available for spray at 1000 gal/min until exhausted; then sump pump will provide 50 gal/min	Radiograph, magnetic particle	2-in. insulation
Pathfinder	None	Radiograph	Insulated on outside; steel picked to be exposed to extreme conditions; air-conditioning units can be used for heat
Humboldt Bay	[Reactor dry well is vented to a water moat]		
BONUS	Spray and quenching moats	Spot radiograph of hemisphere; 100% radiograph of cylindrical walls	Two coats of aluminum paint; limited areas air conditioned; major portion ventilated
Big Rock Point	To be available; not yet specified	Not available	Insulated on outside
Hallam	Not available	Cavity liner spot radiographed	Not available [inner face of cavity liner maintained at not over 150°F by cooling coils]
Enrico Fermi	Not available	Radiograph	Uninsulated
EGCR	None	Not available	Insulated on outside; ventilated
FWCNP	Water spray and gas-to-water heat exchanger recirculates gas in shell	Radiograph	Uninsulated; heat-reflective paint used
Peach Bottom	[None]	[Soap bubble; X-ray as required by code]	[Ventilation air preheat, local unit coolers-heaters as required]
Piqua	Fog-spray fire protection system	10% of spot welds radiographed	3-in. external insulation
CVTR	Not available	All dome welds radiographed	Internal recirculating filtering and air-conditioning system; 122°F (max.), 50°F (min.)

Table IV-1 (Continued)

Short name of plant	Design criteria for containment structure
PWR	Release of main coolant from 14-in. pipe
Indian Point	Release of primary water through a single-ended pipe failure, and release of contents of one steam generator; no cooling considered
Yankee	Release of entire moderator coolant from double-ended 20-in.-diameter pipe failure, and release of contents of one steam generator
N.S. *Savannah*	Instantaneous release of entire primary system contents [and secondary system]
Saxton	All of primary coolant flashes to equilibrium
SPWR	Complete rupture of reactor; all of coolant instantaneously flashes, and a steam pipe ruptures; shell designed for pressure plus temperature rise
VBWR	Instantaneous release of all water and steam in reactor and primary piping; addition of 25% of energy from Zr-H_2O reaction for various cores does not exceed test pressure; no heat transfer to walls considered
Dresden	Instantaneous release of 188 tons of primary water, assuming perfect gas law, perfect mixing, and no heat loss; further calculations under "hot" standby conditions yield pressures of less than 26 psig
Elk River	Release of all primary water; no heat loss; addition of contents of one steam generator would increase pressure by 3.5 psi
Pathfinder	Release of all primary coolant
Humboldt Bay	[Rupture of 12-in. main steam line with 100% water flow to break]
BONUS	Loss of entire coolant; shell designed for hurricanes (150 mph) and seismic load ($c = 0.2$); effect of quenching moat neglected
Big Rock Point	Release of water while reactor is in hot standby condition; 100-mph wind
Hallam	Loss of coolant
Enrico Fermi	Sodium-air reaction
EGCR	Release of all helium and contents of one steam generator
FWCNP	Release of all CO_2 and 50% of stored energy in core; more detailed calculations will be made; shell designed for 150-mph wind
Peach Bottom	100-mph wind load, [0.05 g] seismic load, and 30-psi snow load
Piqua	Steam boiler rupture (3 psig)
CVTR	Release of all coolant, 35% of energy from Zr-D_2O reaction, nuclear excursion of 1×10^6 Btu; no credit for cooling

the components of the primary cooling system in a nitrogen atmosphere in steel-lined shielded cells blanketed by a nitrogen atmosphere, the Hallam designers felt that the sodium-air reaction would be avoided. Thus the Hallam containment consists of the steel-lined cells backed up by a secondary containment afforded by the partially evacuated building structure. On the other hand, the containment structure for the Enrico Fermi plant is designed to withstand the 32-psig pressure from the sodium-air reaction, even though the sodium system components will operate in a low-oxygen environment.

Leakage Specifications

It has been found that conventional steel shells that have quite low leakage rates can be built; for example, ~0.15 per cent per day at the PWR plant, < 0.02 per cent per day at the VBWR plant, and ~ 0.016 per cent per day at the Dresden plant. The design leakage rates for conventional steel shells, however, vary from 0.06 per cent per day at the Enrico Fermi plant to 0.2 per cent per day per pound per square inch gauge at the Piqua plant and up to 320 per cent per day for the outer building at the Hallam plant. For the most part, the specified leakage rates have been based on the exposure at the site boundary or the nearest inhabited area. Normally, the maximum permissible internal and direct exposures at the point of concern are established and then related to the atmospheric conditions and the calculated fission-gas release into the containment structure to establish the allowable leakage rate. Although conservative assumptions are usually made, the results of these calculations are subject to rather large uncertainties, particularly at low wind velocities (see Section V, "Maximum Credible Accident Exposure Doses at Reactor Site Boundaries").

Various methods of leak testing the containment structures are described in the hazards reports. Exhaustive leak-testing provisions are being made for some structures; for example, the steel sphere at the Yankee plant will include a reference-pressure testing system[33,34] that will be available for continuous leakage testing during operation. Prior to operation, the sphere will be tested by the soap-bubble method and, possibly, the Freon or helium leak-detection method. The welds of some structures are to be radiographed (for example, the FWCNP steel pressure vessel); whereas, in contrast, only spot checks of the welds of the Piqua and BONUS containment structures are planned. The leak-testing methods to be used for the various plants are listed in Table IV-1.

Penetrations of Containment Structures

The problems involved in sealing containment structures in the event of an accident, including the sensing of the need for sealing, are numerous. The Indian Point, Yankee, Saxton, N.S. *Savannah*, and CVTR containment vessels will remain sealed during operation. The PWR, SPWR, Dresden, Pathfinder, BONUS, Big Rock Point, Enrico Fermi, FWCNP, Peach Bottom, and Piqua containment shells will be ventilated during operation, but the ventilation ducts will be closed in the event of an accident. Not all the hazards reports describe the properties that will be sensed to activate the ventilation closing valves or the types of instruments required. At Yankee the vapor container is sealed at all times when the reactor is critical or the main coolant system is pressurized. In the SPWR, the ventilation ducts will be closed automatically upon the detection of a high pressure in the building. The Elk River and Piqua shells will be closed by a high activity level; the Pathfinder, by high activity level; BONUS, Peach Bottom, and PWR by high activity level or excessive building pressure; and Big Rock Point, by low reactor pressure, reactor scram signal, high pressure in enclosure, and loss of auxiliary power.

Dependence upon the sensing of a high activity level in the containment structure as the basis for closing ventilation ducts and other penetrations does not take advantage of the time lag between loss of coolant and the onset of fuel-element failure. It would appear that it would be advantageous to reduce the initial pressure rise by venting the relatively uncontaminated gas in the containment shell in the first few seconds after the accident. The penetrations could then be closed automatically when release of activity by the fuel elements was sensed. The degree of pressure reduction that could be obtained by using this scheme would depend on the settings of the activity monitors in the containment shell. If the monitors were set so that the activity normally carried by the coolant would activate the valve-closing mechanisms, the pressure reduction

obtainable would be small. On the other hand, if the activity monitors were set to sense fuel-element failures, it would be essential for the valve-closing mechanisms to operate rapidly. A further disadvantage of dependence on the activity level as the indication of the need for closing the penetrations is that the instruments available for detecting activity are less reliable than those for detecting pressure. It is therefore recommended that two independent properties, such as activity and pressure, be used to activate the mechanisms for penetration closure. As may be seen in Table IV-1, most plants are designed to operate in this manner.

In some plants, activation of the closure mechanisms by spurious signals would not be particularly troublesome, but, in other plants, unnecessary closure of steam lines and other process piping could cause difficulties of such magnitude as to justify multiple sensors and instrument channels. A minimum requirement would be a two-out-of-three system,[35] in which three completely independent channels would convey information to the closure mechanisms. The desired operation could be initiated only by activation of at least two of the channels. Since the probability that two channels would give spurious signals at the same time is small, the possibility of unnecessary operation of the closure mechanisms would be greatly decreased. Also, it would be necessary, of course, to provide for one-out-of-two operation while repairing any one instrument channel. It is the opinion of the reviewer that the problems of instrumentation and control of mechanisms for sealing penetrations of containment vessels are of sufficient importance to require more attention than that indicated by the available hazards reports.

The number of penetrations of the containment structure, if available, is listed for each reactor described in Table IV-1. It is evident that many penetrations are required for instrument lines, process piping, and electrical connections. Not all the penetrating lines are connected to sources of direct or potential contamination, but many are. In calculating exposure doses at positions external to containment vessels, the possibility of one or more automatic valves not sealing the penetration has not usually been considered in the hazards reports (except in the case of the PWR), although the probability that this will happen increases with the number of penetrations that must be closed. In most cases the closure valves will be accessible so that they can be closed manually if the automatic device fails. In some hazards reports, however, this aspect of the problem is not clearly resolved, and no statement is made as to the administrative action to be taken to ensure that the valves are checked and manually operated when necessary.

Shock-Wave Protection

The possible need for shock-wave protection has been discussed only in the hazards reports for the Enrico Fermi and Piqua plants. The containment shell for the Fermi plant was analyzed and found to be of sufficient strength to withstand a force equivalent to 330 lb of TNT with no permanent yield and to 1000 lb of TNT without rupture.[36] The Piqua plant containment vessel is designed to withstand the explosion of the 35,500 mg/m^3 of Santowax-R dust that could be accumulated from small leakage of coolant, although the concentration of the dust is to be maintained at no greater than 94 mg/m^3, which is well below the concentration that constitutes an explosive hazard. Blast effects can be a problem in a pressurized-water system if there is an instantaneous primary system failure as the result of extremely rapid flashing of the water to steam upon decompression.[37] Although not explicitly stated, the large amounts of concrete that make up the biological shields around reactor cores and the factors of safety inherent in the designs of the containment shells provide some measure of blast protection. The effects of blasts on containment shells were considered more fully in a previous issue of *Nuclear Safety*.[38]

Missile Protection

Missile protection is also provided by the biological shield, but in some plants extra shields are required in certain areas. For instance, at the Yankee plant there are 1- and 2-in. steel plates backed by 18 in. of concrete above the control-rod drive mechanism. The VBWR has an 18-in.-thick steel shield above the core when the operating power exceeds 10 Mw. In some cases the steel containment shell is lined with concrete. In other cases the missile problem is dismissed on the premise that the main components are made of ductile materials that are not likely to generate missiles, and the high-speed rotating

machinery is located external to the steel containment shell. In the case of the BONUS plant, the whole system is within the steel hemispherical and cylindrical containment structure, but the turbine unit is enclosed in a missile shield.

Conclusions

The information presented here is incomplete because it is based almost entirely on the hazards reports for the individual reactors. Since some of the reports are preliminary, they cannot be expected to include the detail required for a complete evaluation. Some data from Brittan's report[33] have been used; in fact, Table IV-1 is an expanded version of the table of Brittan's report. As more information becomes available, more meaningful comparisons can be made. The data of Table IV-1 are of interim value, however, in analyzing the various containment concepts. (M. H. Fontana)

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Section V

CONSEQUENCES OF ACTIVITY RELEASE

Dosimetry Aspects of Nuclear Accidents

There have been seven major nuclear accidents since the beginning of the Manhattan Project which have involved significant exposures of personnel to ionizing radiation. These accidents are reviewed here in chronological order, and the sequence of events and conditions of exposure pertinent to an understanding of the physical dosimetry aspects of the accidents are described.

Since the first of these accidents, almost 15 years ago, there has been continuous improvement in the techniques and concepts of radiation dosimetry. For example, dose units such as the gram-roentgen are no longer believed to be adequate indices of radiation exposure, and time has served to accentuate the complexity of the RBE (relative biological effectiveness) concept. An effort has therefore been made in this review to use the more recent techniques of radiation dosimetry to reevaluate the doses that were previously assigned to certain individuals involved in nuclear accidents.

In order to put the physical dosimetry of past accidents on a common and objective basis, the dose to a given individual is specified in terms of the absorbed dose (in rads) which would have been received by a small mass of soft tissue located at the position of the individual. This dose is referred to as the "first-collision dose," consistent with common usage. This definition of first-collision dose is not, however, consistent with the mathematically convenient definition given by Snyder.[1] His definition refers to the energy per gram imparted to a small-volume element of tissue located at a given depth below the surface of the body by neutrons that have made no collisions until they reach the point in question. Since correlations of biological data with a single dose reading are far from perfect unless physical conditions of the exposure other than the energy imparted are specified, as much information as is available on the parameters of each accident has also been considered. First-collision doses are given, for example, in terms of rads to soft tissue for neutrons and gamma rays separately. This procedure is consistent with the desirability of establishing RBE factors for acute irradiation of man based on accident data.

Measurements of the formation of Na^{24} in the body via the reaction $Na^{23}(n,\gamma)Na^{24}$ have been found[2] to give the most reliable evaluation of the neutron dose to persons exposed in mixed-radiation fields. The neutron dose to the body is related to the spectrum of incident neutrons, which is determined from threshold-detector measurements. This method of dose measurement has been termed "spectral" dosimetry.[3] The relation between the Na^{24} formation in the body and the neutron spectrum is simple, and, if the neutron spectrum escaping from a critical assembly can be estimated or measured, the Na^{24} activation in a man may be interpreted in terms of the first-collision dose at his position. This method has been used here for the reevaluation of the doses involved in accidents for which data on spectral distribution of neutrons were available or could be estimated. Estimates of the gamma-to-neutron dose ratio have been used for assigning both gamma and neutron doses on the basis of Na^{24} activation in the body.

Dose estimates have been made only for cases in which the radiation exposure was acute, external, penetrating, and generalized. Doses from alpha and beta radiation, either internal or external to the body, have been neglected because they are thought to be relatively unimportant in the usual criticality accident.

Los Alamos Accident LASL-I

The first nuclear accident occurred[4] at the Los Alamos Scientific Laboratory, Los Alamos, N. Mex., in August 1945. During a criticality experiment involving a metal system, there was a nuclear excursion, and two people were exposed to the resulting mixed-radiation field. Case LA-1, who died following the accident, was touching the assembly when the chain reaction was initiated, and he received a very high dose to his hands and a smaller dose to the bulk of his torso. He was exposed to considerable gamma radiation from fission products while dismantling the experimental system, and he would possibly have survived had he left the room immediately. His actions during this period are not well known, and the total dose he received is correspondingly uncertain. Neither he nor the other exposed person was wearing a film badge at the time of the accident. Since the number of fissions that occurred in the assembly was not known accurately, the best index of his neutron exposure was the Na^{24} activity induced in his blood.

As stated above, the Na^{24} activity in the blood may be related to the number of incident neutrons if the energy distribution of the neutrons is known. In order to determine the first-collision dose in this case, the neutron spectrum of the excursion was assumed to be similar to that of the Godiva reactor, an unreflected (bare) reactor[5] studied at Los Alamos in 1956. It was further assumed that the accident spectrum did not vary with distance over the range at which the exposure occurred. It is known that the attenuation of the tamper around the critical experiment was not high.[6] The relation between Na^{24} activity (in disintegrations per second per milliliter of blood serum) and the spectral distribution given by Godiva threshold-detector measurements (Table V-1), as corrected,[7] was then used to establish the neutron dose correlation, i.e.,

$$D_n = 5.0K$$

where K is the activity of the blood serum in disintegrations per second per milliliter, D_n is the neutron dose in rads, and 5.0 is the relation between the Na^{24} activity and the neutron spectral distribution.[3] It is known that the quantity of Na^{23} in blood serum is remarkably constant in man at the value of 3.2 mg/ml (see reference 1), and this fact has been used in converting from activity per unit mass of Na^{23} to activity per unit volume of blood. A neutron dose of 288 rads was obtained for this case, based on the Na^{24} value for Case LA-1 given in reference 8. The data for this case and the other Los Alamos accidents discussed below are summarized in Table V-2.

Table V-1 FAST-NEUTRON LEAKAGE SPECTRUM FROM GODIVA REACTOR

Detector	Relative flux*
Pu^{239} (2-cm B^{10}) (n,f)	1.00
Np^{237} (n,f)	0.86†
U^{238} (n,f)	0.46
S^{32} (n,p)	0.21

*Measurements were made in August 1956. At 1 meter from the center of the Godiva reactor, the actual plutonium flux was 3.0×10^{10} neutrons/cm^2 for 2.04×10^{15} fissions.

†This value includes a correction for the new fission cross-section value for Np^{237} of 1.6 barns, based on a personal communication from R. B. Murray, Oak Ridge National Laboratory (see also reference 7).

Case LA-2, who was exposed at a greater distance from the reactor than Case LA-1, did not remain in the vicinity after the excursion. Again, based on measurements of the gamma-to-neutron dose ratio for leakage from the Godiva reactor, it is estimated that the gamma-to-neutron dose ratio was approximately 1:10 for Case LA-2 in the LASL-I accident. It was assumed that the ratio was independent of distance from the source for the distance involved. The estimated first-collision dose for Case LA-2 is given in Table V-2.

Los Alamos Accident LASL-II

The circumstances surrounding the second major radiation exposure at Los Alamos, which occurred on May 21, 1946, were quite similar to those of the first accident. Personnel film-badge data were not available, and data from badges located on the walls of the room were not very helpful. Eight persons were exposed to significant radiation doses in this incident, and Case LA-3 died as a result. It was felt that none of the victims received uniform total-body exposures because of varying degrees of partial body shielding.[4]

The neutron doses to the exposed persons have been estimated, as in the case of the LASL-I accident (i.e., by using neutron spectral

Table V-2 ESTIMATED FIRST-COLLISION DOSES* TO PERSONS EXPOSED IN LOS ALAMOS NUCLEAR ACCIDENTS

Accident	Reference	Case	Na^{24} activity, dis/(sec)(mg)	K, dis/(sec)(ml)	Neutron dose (5.0K), rads	Gamma dose, rads	Total dose, rads
LASL-I	8	LA-1	18.0	57.6	288	Uncertain	
		LA-2	1.10	3.5	18	2	20
LASL-II	9	LA-3	74.8	239.0	1190	120	1310
		LA-4	13.2	42.2	213	21	234
		LA-6	7.1	22.7	117	12	129
		LA-7	3.0	9.6	48	5	53
		LA-8	2.0	6.4	32	3	35
		LA-9	1.54	4.9	24	2.4	26
		LA-10	1.22	3.9	19	2	21
LASL-III	20	LA-11		451	2200	~6600	8800

*These values are based on Na^{24} activity in blood serum.

data obtained from the Godiva reactor, together with Na^{24} activation data for the blood of the patients).[9] The doses thus derived are presented in Table V-2. It was assumed, as for Case LA-2, that the gamma-to-neutron dose ratio was 1:10 for all those exposed. This is probably a reasonable assumption, except for Case LA-9, who returned to the scene of the accident twice after his initial exposure.

Argonne National Laboratory Accident

The third reactor criticality accident occurred[10] at the Argonne National Laboratory, Argonne, Ill., on June 2, 1952. Four persons were exposed to a mixed-radiation field in which the gamma-to-neutron dose ratio was about 10:1. The critical assembly consisted of a quasi-cylindrical metallic core surrounded by a natural-water moderator in a plastic tank. There were no fatalities.

Cases A-1 to A-3 were wearing film badges which were so blackened that it was necessary to evaluate the doses in terms of the reduced silver left after development of the film. The gamma dose to patient A-4 was evaluated from a knowledge of his position and times of exposure to prompt and delayed fission gamma rays. The gamma dose is given by Hasterlik and Marinelli[10] in terms of first-collision dose, i.e., the correction for the effect of the body on the dose has been made. The doses were found to be fairly constant over the bodies of those exposed and could be considered as uniform whole-body irradiation.

Threshold-detector measurements of the neutron spectrum escaping from this assembly are not available. However, Hasterlik and Marinelli quote accurate neutron dose values determined with proportional-counter dosimeters.[11] These were obtained by measuring the ratio of the fast-neutron dose to gamma dose during an experimental run of the reconstructed reactor and applying this ratio to the film-badge readings. The resulting dose values are shown in Table V-3.

Table V-3 ESTIMATED FIRST-COLLISION DOSES TO PERSONS EXPOSED IN ARGONNE NATIONAL LABORATORY NUCLEAR ACCIDENT

Employee	Gamma dose,* rads	Neutron dose,† rads	Total dose, rads
A-1	145	14.2	159
A-2	116	9.6	126
A-3	55	5.5	60.5
A-4	10	0.8	10.8

*These values were obtained from readings of the reduced silver left after development of the film badges; corrected for buildup in air and for body shielding.

†Based on gamma-to-neutron dose ratio and film-badge readings; values are precise to ~10 per cent, based on dosimetric measurements obtained during reconstructed accident.

USSR Accident

At some undisclosed time prior to July 1956, a reactor accident occurred in the USSR which caused "short, general, external" exposure of two persons to neutrons and gamma rays. The

clinical results of the exposures have been described by Guskova and Baisogolov.[12] They assigned doses of 300 and 450 "r" to the exposed people. The gamma-to-neutron ratio in the radiation field is not given in their report, however, and no information concerning the structure and composition of the reactor is presented on which to base an estimate of the ratio Furthermore, the methods of dosimetry used are not described, and the dose assignments must therefore be regarded as ambiguous.

Oak Ridge Accident

A criticality accident occurred on June 16, 1958, at the Y-12 Plant at Oak Ridge, Tenn.[2,13] Criticality was reached when an aqueous solution of an enriched uranium compound was drained inadvertently into a waste drum having dimensions which permitted a critical mass to be formed. Eight persons received general whole-body radiation doses, and five of the eight received total doses in excess of 200 rads. There were no fatalities.

None of those exposed to the radiation field were wearing film badges, and therefore blood sodium radioactivity was chosen as the most reliable measure of the neutron dose in this accident. The neutron dose to a man at a given distance from the assembly was deduced from the Na^{24} concentration in his body by employing a mockup reactor and a "phantom" (in this case, a burro). The phantom was exposed to the neutron field escaping from a mockup reactor of dimensions and composition similar to those of the accident assembly. The Na^{24} concentration in the blood serum of the phantom after exposure to a measured first-collision neutron dose was determined and used to convert the Na^{24} activity data for the exposed persons to first-collision neutron doses, assuming that the neutron spectrum did not vary with distance from the reactor. During another run of the same mockup reactor, a measurement of the gamma-to-neutron dose ratio was made for use in determining the gamma doses. Corrections were made for the different contributions from fission-product gamma rays in the two cases. Calculations[2] based on the neutron and gamma-ray leakage spectra from the mockup reactor gave dose estimates that agreed well with the values based on the Na^{24} activity of the individuals. The doses of the exposed persons are given in Table V-4.

Table V-4 ESTIMATED FIRST-COLLISION DOSES* TO PERSONS EXPOSED IN OAK RIDGE Y-12 NUCLEAR ACCIDENT

Employee	Neutron dose, rads	Gamma dose, rads	Total dose, rads
OR-1	96	269	365
OR-2	89	250	339
OR-3	86	241	327
OR-4	71	199	270
OR-5	62	174	236
OR-6	18	50.5	68.5
OR-7	18	50.5	68.5
OR-8	6	16.8	22.8

*These values are based on Na^{24} activity in blood serum.

Yugoslavian Accident

In an accidental supercritical excursion of a zero-power reactor on Oct. 15, 1958, at the Boris Kidrich Institute of Nuclear Sciences in Vinca (near Belgrade), Yugoslavia, eight persons received doses of neutrons and gamma rays above the maximum permissible concentration.[14] One man died as a result of the irradiation.

The reactor was a natural-uranium heavy-water unreflected assembly in a cylindrical tank approximately 2 meters high and 2 meters in diameter. The exposed persons were located in the range of 3 to 8 meters from the reactor at the time of the excursion.

Although incomplete information was available concerning the spectra of neutron and gamma leakage from the reactor and the disposition of people and apparatus at the time of the accident, it was possible to make estimates of the doses from Na^{24} activation data for the six patients treated in Paris at the Curie Foundation.[15] One account of the accident[16] stated that it was known, from both calculations and the experiment, that the amounts of thermal- and epithermal-neutron leakage from the reactor were equal. It seems reasonable therefore to take the fast-neutron leakage spectrum from the reactor to be roughly similar to that measured in the Y-12 accident study.[13] Hence it has been assumed that the total neutron spectrum in the Yugoslavian accident was equal to that in the Y-12 case for epithermal energies (>5 kev) and that the thermal-neutron flux was equal to the total flux above 5 kev. Based on these assumptions and the spectral dosimetry method[3] described above, the neutron dose correlation was found to be

$$D_n = 3.92A$$

where D_n is the neutron dose in rads and A is the total measured Na^{24} activity in the body in microcuries. The values of A for the various patients and the estimated doses they received are given in Table V-5. Values are also given in Table V-5 for the concentration, K, of the activity in the blood serum, based[17] on the amount of Na^{23} in each individual being 105 g.

Table V-5 ESTIMATED FIRST-COLLISION DOSES* TO PERSONS EXPOSED IN YUGOSLAVIAN ACCIDENT

Case	Total Na^{24} activity† (A), μc	K,‡ dis/(sec)(ml of blood serum)	Neutron dose,§ rads	Gamma dose,¶ rads	Total dose, rads
V	82	92.5	320	320	640
M	75	84.6	290	290	580
G	76	85.7	300	300	600
D	63	71.0	250	250	500
H	53	60.0	210	210	420
B	45	50.8	175	175	350

*Based on the total Na^{24} activity in the body of the exposed person.

†Total Na^{24} activity (A) in exposed individual; data are taken from reference 15.

‡Based on 70-kg standard man containing 105 g of Na^{23} (see reference 17); $K = 1.13A$.

§Based on fast-neutron spectrum being the same as for the Y-12 accident and there being one thermal neutron for each fast neutron; $D_n = 3.47K$.

¶Based on gamma-to-neutron dose ratio of 1:1 (see references 16 and 18).

The gamma doses given in Table V-5 were deduced from a reported measurement[14,16,18] of a gamma dose of 1 mr/hr at a thermal-neutron flux level of 155 neutrons/(cm²)(sec) around the reactor when operated in earlier experiments. From this value and measurements of the neutron-to-gamma dose ratio around similar reactors, it was estimated that the neutron-to-gamma dose ratio was approximately 1:1.

Los Alamos Accident LASL-III

The most recent serious exposure of a person to radiation occurred on Dec. 30, 1958, at the Los Alamos Scientific Laboratory.[19] In the course of a routine plutonium salvage operation, the material attained a critical state. Case LA-11 (see Table V-2) was standing in close proximity to the critical system and received a lethal dose.

If it is assumed that the neutron and gamma-ray fields were comparable with those measured in the Y-12 mockup experiment,[13] the neutron dose may be approximated as

$$D_n = 5.2K$$

The value of K was found[20] to be 451 dis/(sec)(mg). This relation between D_n and K is nearly the same as that derived from the Godiva measurements. Based on this relation, the fast-neutron dose received by Case LA-11 was 2200 rads, and, using a gamma-to-neutron ratio of 3:1, the gamma dose was 6600 rads, as indicated in Table V-2.

The foregoing analyses of the seven major accidents show clearly that the dosimetric aspects left much to be desired. The need for accurate data in an accident situation is superseded only by the need for a rapid determination of the dose. Careful considerations of these needs led to the recommendation of the dosimetry system reviewed in the following subsection.

(R. H. Ritchie)

System for Dose Measurements in Radiation Incidents

The nuclear accidents that have resulted in significant radiation exposures of people have been characterized by the lack of sufficient monitoring instruments and badges to give adequate indications of the doses received. In most cases it has been necessary to mock up the accident in order to obtain the information required for estimating the doses, and therefore the dose estimates have not been available until several days after the incident. A study by members of the ORNL Health Physics Division of the problems involved in obtaining dose measurements has led to the development of a dosimetry system that will give the dose to any individual, to within ±15 per cent, in a matter of a few hours.[3] The system is designed (1) to measure gamma-ray doses in the presence of neutrons, (2) to measure neutron doses in the presence of gamma rays, (3) to give information on the orientation of the exposed person, and (4) to give immediate indications of the persons exposed so that they may be separated from the unexposed persons.

Description of System

The equipment required to give this comprehensive picture of accident doses is too com-

plicated and too large to be worn by each person, and therefore part of it must be placed at fixed locations within laboratory areas where an incident is possible. Such "dosimetry stations" are of two types: complete stations and secondary stations. A complete station consists of a series of threshold detectors capable of measuring the spectra of fast and thermal neutrons and a device for measuring the gamma dose in the presence of neutrons. If there is a radiation shield between the possible point of excursion and the personnel who might be exposed, two complete dosimeter stations are required, one on each side of the shield. The secondary stations contain, in particular, sulfur and gold samples for thermal-neutron-flux measurements, and they are placed at numerous points throughout the area involved.

The personnel in the hazardous area are required to wear beta-gamma film badges of the type developed by Davis et al.,[21] which also contain sulfur and gold foils for neutron-activation measurements. In addition, each individual must wear a belt containing gamma-ray and neutron dosimeters at fixed intervals in order to provide a means for determining his effective orientation with respect to the source. The dosimeters in the belt are glass rods for gamma-dose measurements and sulfur and gold samples for neutron-flux determinations. Both measurements are important because the effective gamma-ray and neutron doses may be different for different orientations.

Principles of Dose Determinations

The data obtained from this dosimetry system, together with the Na^{24} activity in the blood of an exposed individual, provide the information needed for a determination of the neutron dose. The probability of neutron capture by the sodium in the blood has been shown to be roughly constant throughout the fast-neutron-flux spectrum and to be about three times the probability of capture of incident thermal neutrons.[2] Since the ratio of thermal-to-fast neutrons may vary considerably with position, the values obtained for activation of the blood sodium are adjusted to take into account the effect of thermal neutrons as required on the basis of the data from the gold samples in the badge or belt. The spectrum of the fast neutrons does not vary greatly with distance or position, and thus the neutron dose can be determined by using the neutron dose calculated from the threshold detectors at the dosimeter station and by normalizing the flux obtained from the sulfur sample at the dosimeter station to that obtained for the person from the sulfur sample in his radiation belt or film badge. The neutron dose can also be determined by correlating the blood sodium activation measurements with direct calculations of the fast-neutron leakage spectrum, as described in the preceding review.

The gamma-ray dose can be determined from the ratio of the gamma-to-neutron dose found at the dosimeter station. Since this ratio does not vary much with position, the gamma dose to the individual can be obtained by applying this ratio to the neutron dose he received. In some cases an estimate can be made by calculating the gamma-to-neutron dose ratio from the source and applying this ratio as in the first method.

The orientation of the individual can be determined approximately by using data obtained from the radiation belt. An individual may be facing the point of excursion and may receive his neutron dose from the front; if he then runs away, he will receive part of his gamma dose through the back. Measurements made at the ORNL Critical Experiments Facility using phantoms have indicated that the fast-neutron flux, as measured by the sulfur threshold detector, differed by a factor of 10 from front to rear of a person.[22] The thermal-neutron-flux and gamma-ray doses, as measured by gold and glass dosimeters, differed by a factor of approximately 2 from front to rear.

Any immediate indicator of exposure must be able to show either gamma-ray or neutron dosage. Gamma-ray exposure may be detected, for example, with a chemical dosimeter in the badge that changes color immediately upon irradiation. Activation of indium and gold samples in the badge would indicate neutron irradiation; indium is the more sensitive, but gold has a longer half life.

Mechanics of Dosimetry

The threshold detector developed at ORNL provides for measurements of neutron flux in five energy regions.[23,24] The thermal-neutron flux is determined by obtaining the difference in activation of gold alone and of gold surrounded by cadmium. The fast-neutron flux is determined from the fission reactions in Pu^{239}, Np^{237}, U^{238}, and the (n,p) reactions in S^{32}, which provide

measurements of the flux above 1 kev, 0.75 Mev, 1.5 Mev, and 2.5 Mev, respectively. From these measurements, the total neutron dose can be obtained. Since Pu^{239} has a high thermal-neutron fission cross section, the thermal neutrons must be removed in order to make it a suitable threshold detector. In the case of Np^{237}, a competing high thermal-neutron capture cross section gives Np^{238}, which emits a 1.0-Mev gamma ray with a 2.3-day half life. In addition, U^{238} usually contains a small amount of U^{235}, which has a high thermal-neutron fission cross section. Therefore it is advantageous to shield all the fission foils from thermal neutrons. A B^{10} shield with a thickness of 2.0 g/cm^2 is sufficient to do this.

All the foils are counted on scintillation counters, and thus only one set of electronic equipment is required. The fission foils are usually counted first because the fission activity decays rapidly. The recommended apparatus[3] consists of two 4-in.-diameter 2-in.-thick thallium-activated sodium iodide scintillators on each side of the fission foil to be counted. In addition, a $\frac{5}{8}$-in.-thick lead filter is placed between the foil and scintillators to reduce the high soft-gamma-ray background, especially in the case of Np^{237}. In order to further reduce the background, the pulses counted are those above a discriminator level of 1.2 Mev. The decay patterns of the three fission foils are somewhat different, i.e., Np^{237} decays the slowest and Pu^{239} decays the fastest, and such differences must be taken into account. A $1\frac{1}{2}$-in.-diameter 1-in.-thick NaI(Tl) scintillation counter is adequate for counting gold foils, and with a 2.7-day half life there is no urgency about the determination. In the case of sulfur, the $S^{32}(n,p)P^{32}$ reaction allows only the beta-ray emissions of P^{32} to be counted. A $1\frac{11}{16}$-in.-diameter $\frac{1}{16}$-in.-thick plastic scintillator is used to count the beta rays emitted by the P^{32} in the $1\frac{1}{2}$-in.-diameter $\frac{3}{8}$-in.-thick sulfur pellet. If there is insufficient activity in the sample for counting in this manner, the sulfur pellet is burned in a 0.001-in.-thick aluminum-foil dish. This treatment removes the sulfur and leaves the P^{32} on the aluminum. The sides of the dish are then folded down, and the dish is counted with a lead reflector on the opposite side of the foil from the scintillator. Since this treatment removes the self-absorption of the sulfur, the counting rate is increased by a factor of 19 for the size of pellet used. Since the half life of P^{32} is 14.3 days, there is no urgency in the burning or counting of the sample.

For counting Na^{24} activity in blood samples, the larger of the scintillation counters mentioned above may be used; one crystal is adequate. The amount of blood required depends on the neutron dose. If the dose is suspected to be over 100 rads of neutrons, only 10 ml of blood serum is needed. If the dose is less than 10 rads, 100 ml should be taken. It is convenient to count only the blood serum that may be removed from whole blood by allowing it to clot and then centrifuging. Standard polyethylene boxes are used to contain the samples. If the samples are counted less than 4 hr after exposure, some activity due to Cl^{38} (half life of 37.5 min) may be found, but this activity presents no difficulty because the half life of Na^{24} is 15 hr. For planning purposes, it may be assumed that, in the case of fission neutrons, the number of disintegrations of Na^{24} per second per milliliter of blood serum is equal to one-fifth of the neutron dose in rads.

Chemical dosimeters applicable over a gamma-ray dose range from approximately 5 to 2×10^6 rads can be made. Such a system may consist of tetrachloroethylene (TCE), which contains a small amount of free-radical chain-reaction retarder and which is overlaid with an aqueous phase containing a *p*H indicator. Under radiation these chemical chain reactions liberate water-soluble acids in an amount directly proportional to the dose in a given range. The response caused by liberation of the acid produced can be evaluated by changes in conductivity of the solution or by changes in *p*H, as measured electrometrically or colorimetrically. The TCE two-phase system produces approximately 3 to 4 per cent as much acid from a rad of neutrons as that produced by a rad of gamma radiation. Because of the lack of hydrogen, the TCE system absorbs a relatively low amount of energy from the fast-neutron portion of a mixed-radiation field. When the dosimeter is housed in a lithium-tin shield, it is energy-independent to within ±8 per cent from 90 kev to 10 Mev. By using Chlor Phenol Red as an indicator and by comparing the light transmission at wavelengths of 580 and 432 mμ, as measured with a spectrophotometer, with the light transmission of a standard, the gamma-ray dose can be determined to within ±3 per cent. The chemicals should be inspected for deterioration and possibly replaced every nine months.

Metaphosphate glasses containing silver may be used to measure gamma-radiation doses in the range of 5 to several thousand rads. The glass, in the form of small glass rods (1 mm in diameter and 6 mm long), and a convenient reader are commercially available. When the glass is exposed to ionizing radiation, loosely bound electrons are freed from negative ions. The freed electrons migrate through the glass, and some of them are trapped by interstitial silver ions to form a special type of F-center. When the F-center, or photoluminescent center, is excited by ultraviolet light, it returns to the ground state by emitting a photon in the red region (6400 A). Since the response increases for several hours after exposure, the glass rods are usually read after 24 hr. A 0.020-in.-tantalum, 0.015-in.-Teflon filter is sufficient to reduce the energy dependence to within ±10 per cent above 40 kev. The response of the glass rods to fast neutrons is negligible compared with the response to gamma rays. The response to thermal neutrons is sufficiently high to necessitate encapsulation in lithium cans for accurate measurement of gamma-ray dose in a thermal-neutron field. The small dimensions of the glass justify the use of lithium enriched with Li^6. A shield of $\frac{1}{8}$-in.-thick 96 per cent Li^6 material is sufficient.

The recently modified ORNL badge provides for two dosimeter films, with plastic and metal sheets as filters. A 0.5-g pellet of sulfur and a $\frac{7}{16}$-in.-diameter 0.005-in.-thick gold foil are included in a cavity in a slide, and a $1\frac{1}{4}$- by $\frac{1}{4}$- by 0.015-in.-thick indium foil perforated with an identification number is located in the front of the badge. Chemical and/or silver phosphate glass dosimeters can be added as desired. The Du Pont film packet 553, which has a hard-gamma-ray dose range of 50 to 50,000 millirads, is used in conjunction with the filters for routine beta-gamma dosimetry. The Eastman NTA film, which has a range of 20 to 10,000 millirads for 2-Mev neutrons, is used for routine neutron dosimetry. Chemical or glass type dosimeters are included in the badges of individuals who work in areas where a radiation incident that would cause a high gamma-ray exposure could occur. Sulfur, gold, and cadmium detectors are installed at dosimeter stations to provide for neutron dose determinations. The indium and gold foils in the badges would provide a quick method of identifying the individuals with high neutron exposures in the event of a nuclear accident, and the chemical dosimeters, which would change in color from red to yellow upon receiving a gamma dose of 25 rads, would indicate those individuals with high gamma-ray exposures.

As mentioned above, this dosimetry system evolved from studies of problems that were encountered following radiation accidents. The minimum recommended system[3] for use in areas where a nuclear incident might occur consists of threshold detectors, chemical and/or glass dosimeters, and facilities for counting Na^{24} activity in blood serum. If available, the film-badge package and radiation belt should be added. (F. J. Davis)

Ocean Disposal of Radioactive Waste

The contamination of water resources with radioactive waste effluents presents complex problems. Since there is limited quantitative information on the physical, chemical, and biological processes occurring in aquatic environments, the disposal of radioactive waste by discharge into bodies of water must be governed by seemingly conservative regulations. As the ecologist supplies additional information on the cycling and accumulation of radionuclides in such environments, the regulations may be modified to permit more complete utilization of the diluting and dispersing capacities of water bodies.

The recent Congressional hearings[25] on industrial radioactive waste disposal and the discussions at the International Scientific Conference on the Disposal of Radioactive Wastes[26] emphasized the grave concern that exists with respect to the suggestion that radioactive waste be discharged into the ocean. During the first five days of the Congressional hearings, there was little public concern; however, on the day that the disposal of low-level radioactive waste into the ocean was discussed, 13 members of Congress appeared before the Joint Committee on Atomic Energy, or submitted statements to it, protesting the sites suggested for disposal of packaged low-level waste in the report of the National Research Council (NRC) of the National Academy of Sciences (NAS). In fact, that particular session of the hearings was scheduled because of public reaction to the report "Radioactive Waste Disposal into Atlantic and Gulf

Coastal Waters,"[27] which presents the results of a study of the feasibility of using selected areas in the Atlantic Ocean and Gulf of Mexico coastal waters as receiving grounds for the disposal of packaged low-level radioactive waste. The limiting rate of disposal was set so that the physical and biological processes in the ocean would not return the radioactivity to man at a rate above the maximum permissible rate of ingestion of a given radionuclide in drinking water. Based on this limitation, 28 sites removed from submarine cable operation and fishing activities in the Atlantic and Gulf of Mexico coastal areas were recommended, provided that a comprehensive study of the local circulation and the biota confirmed the feasibility. Coastal estuaries and bays and the regions immediately seaward of these areas were specifically rejected because shoreward transport along the bottom in those regions would tend to intensify the rate of return of a contaminant to man.

The rate of disposal of radioactive waste in the coastal waters was based upon the maximum permissible concentrations of radioactive isotopes in sea water which will not lead to hazardous exposure of a human whose sole source of protein is fish, taking into account the capacity of marine organisms to accumulate various isotopes above the level of their environment.

In addition, the NRC report[27] recommended that no more than 250 curies of Sr^{90} or the equivalent of other isotopes be discharged to any one disposal area in one year, with a maximum monthly limit of 100 curies; the waste was to be floatproof and to be discharged in floatproof containers. Adjacent disposal areas were to be separated by at least 75 miles, with a maximum of three disposal areas in a 300-mile section of the coastline, unless exchange between disposal areas was impossible. Finally, the disposal site was to be periodically monitored to measure the spread of the radioactivity.

The major Congressional criticisms of the report were that some of the sites recommended for study were as shallow as 42 ft (Site 11a, which was to serve Galveston, Tex.) and as close to shore as 10 miles (Site 2a, which was to serve Providence, R. I.) and that the sites were selected without preliminary study. Most of the criticisms ignored the fact that these were only suggested sites that were subject to change prior to disposal operations after detailed examination. The Pacific Coast has not been neglected as a disposal site, but the report on radioactive waste disposal into Pacific coastal waters has not yet been released.

U. S. Ocean Disposal Practices

To date the United States has disposed of approximately 8000 curies of packaged radioactive material into the Atlantic Ocean, as well as the reactor structure of the *Seawolf* prototype, and approximately 14,000 curies of packaged radioactive material into the Pacific Ocean.[28] No disposals have been made into the Gulf of Mexico.

The U. S. coast line is 4840 miles long,* and, if a disposal area 50 miles seaward from the coast and a total fallout of 40 millicuries of Sr^{90} per square mile (average fallout over the continental United States as of October 1958) are assumed, approximately 10,000 curies of strontium alone has already fallen on the 50-mile-wide area of ocean contiguous to our shores. Since Sr^{90} has a yield of only 5.8 per cent, it is obvious that considerably greater quantities of radioactive materials have fallen on our coastal areas than have been purposefully discharged there.

Disposal of high-level radioactive waste has been limited to storage under controlled conditions. Work by Bogorov and Kreps[29] and by Piccard and Walsh[30] has shown that even the ocean trenches at depths of over 30,000 ft have organisms and circulation. Bogorov and Kreps state that radioactive waste compounds dumped into an ocean trench and dissolved will inevitably be brought to the upper layers of the ocean in regions exploited by man and that sooner or later they will be absorbed by vegetable and animal organisms, fishes, and mammals. Hence there are no natural regions for isolated storage in the ocean, and only low-level waste can be safely discharged.

Utilization of the Sea for Waste Disposal

Although there are large gaps in the knowledge of the physics, chemistry, and biology of the oceans, enough information is available for making conservative estimates of the safe amounts that can be discharged. Perhaps the

*Length of general shore line, as given by the U. S. Coast and Geodetic Survey.

most comprehensive compilation of oceanic information is contained in *The Oceans*.[31] A more recent volume is devoted solely to the effects of atomic radiation on oceanography and fisheries.[32] This work details present knowledge of the fate of fission products discharged to the ocean and points out the areas where additional research is required. It is stated[32] that more information is needed on (1) water movement as a result of convection and eddy diffusion, (2) the role of the ecological system in the transport of elements in the sea, (3) the precipitation of fission-product elements by physical, chemical, and biological processes, and (4) the uptake, accumulation, and loss of radionuclides by aquatic organisms.

Based on the present knowledge of the fate of fission products in the marine environment, another working group of the NRC has reported on considerations relative to the disposal of radioactive waste from nuclear-powered ships into the marine environment.[33] The purpose of this report was to provide an evaluation of the potential capacity of the marine environment to receive certain radioactive waste materials originating from normal operations of nuclear-powered ships having water-cooled reactors. It was assumed that sea disposal should contribute no more than one-third of the maximum permissible intake of radioactive material, and the operation of 300 ships powered with 60-Mw reactors was considered. The conclusions of the study were that (1) no waste should be discharged in harbors, estuaries, and coastal waters out to 2 miles from the shore line; (2) in the coastal area from 2 to 12 miles from the shore line, liquid waste could be discharged if no single discharge contained more than 0.5 curie of radionuclides with half lives greater than 6 hr; (3) on the outer continental shelf from 12 miles offshore to the 200-fathom contour line, spent ion-exchange resins could be discharged in batches of up to 50 curies into nonfishing areas if the ship were under way and more than 3 miles from a neighboring vessel and if the spent resins were denser than sea water; (4) beyond the 200-fathom contour line, spent ion-exchange resins could be discharged in batches of up to 500 curies if no ship made more than one such discharge per month and if no such discharges were made in the major oceanic fishing areas. The report emphasizes that no solid or spent resins should be discharged into known fishing areas. In addition, the report strongly recommends that (1) a systematic monitoring program be established to determine the consequences of the release of radioactive waste from nuclear-powered ships into the ocean and (2) records of such release be maintained.

It should be emphasized that the NAS-NRC working groups only report and make recommendations to the AEC; it is the responsibility of the AEC and, ultimately, Congress, to decide the exact policies concerning the discharge of radioactive waste into the ocean. Although many nations are considering or planning ocean disposal of radioactive waste,[34–38] only Great Britain is presently engaged in the continuous disposal of liquid waste into the ocean.[39] From June 1953 to May 1958, approximately 200,000 curies of gross beta and 250 curies of gross alpha activity had been discharged from Windscale into the Irish Sea. Included in the 200,000 curies of beta activity were approximately 5000 curies of Sr^{90}. Prior to the discharge of radioactive waste, dye was introduced into the general area to determine the dispersion that would prevail.[40] Based on these studies, it was shown that the main hazard would be from the radioactive materials in edible fish and seaweed and from the radiation field of the shore. Other risks were shown to be of a lower order of importance. The preliminary studies, assuming a safety factor of 10, indicated that it would be feasible to discharge about 100 curies of beta activity and 0.1 curie of alpha activity per day.[41]

In 1952 and the first quarter of 1953, experimental discharges at a lower rate were made to check the dispersion assumptions and the concentrations in the fish and seaweed and on the shore. The studies showed that equilibrium conditions were achieved after discharges lasting for only a few months. The short time for equilibrium to be achieved indicated a fairly rapid transfer of activity out of the immediate environment, including the sea bed.

From the results of releases continuing up to April 1954, a release of 1000 curies per day was allowed, with restrictions on Sr^{90} and Ru^{106} content.[42] Results from releases since that time indicate that 20,000 curies per month can be safely released and that it may be possible to release as much as 100,000 curies per month.[39]

Any discharge to the oceans, whether high- or low-level, must be made with the realization that there are strong objections to the practice.

The United Nations Conference on the Law of the Sea[43] unanimously resolved that:

> (1) Every state shall take measures to prevent the pollution of the seas from the dumping of radioactive wastes, taking into account the standards and regulations which may be formulated by the competent international organizations; and (2) all states shall cooperate with the competent international organizations in taking measures for the prevention of pollution of the seas or air space above resulting from any activities with radioactive materials or other harmful agents.

It is obvious that discharge to the ocean should be made only after careful scientific study and under extremely conservative conditions.

(F. L. Parker)

Biological Concentration of Radioisotopes from Aquatic Environments

The release of liquid low-level radioactive wastes to fresh-water streams occurs either intermittently or continuously at the various AEC installations. Such releases of radioactive materials are regulated so that the maximum permissible concentrations in drinking water (MPC_w) for human consumption, as established by the International Commission on Radiological Protection (ICRP) and the National Committee on Radiation Protection (NCRP) are not exceeded. It has been observed,[44,45] however, that the MPC_w regulations consider only drinking-water standards, whereas additional effects may result from concentration of isotopes in aquatic food chains. Aside from the possible deleterious effects on the aquatic organisms, there is a potentially serious hazard from the addition of radioactive isotopes to humans by an indirect route,[46] i.e., through consumption of fish or other aquatic or semiaquatic organisms.

The problem in aquatic habitats is not entirely biological in that physical and chemical factors are also of importance. Furthermore, it is known that the relative importance of these factors varies in different aquatic habitats. It has been shown that fission-product waste materials from ORNL undergo selective adsorption on clay minerals in the bottom sediments of the Clinch River.[47] The fission products Cs^{137} and Ce^{144} are readily adsorbed, whereas Sr^{90} remains in solution. The uptake of Cs^{137} by emergent vegetation is greater for plants rooted in sand or gravel than for plants rooted in mud.[48] *Polygonum persicaria L.* (smartweed) raised in soil in a simulated shallow-pond environment accumulated 450 times as much Cs^{137} as plants grown on an irrigated-field environment.[49] Cesium released to a stream with a sand, gravel, or rock bottom is not adsorbed as strongly on sediment particles and therefore is more readily available for uptake by aquatic organisms than the cesium adsorbed by the finer particle muds. With continual releases of fission products, the radioactivity in the bottom sediments will increase until an exchange equilibrium is reached with the supernatant water. River currents bring in more sediment from upstream and disperse radioactive sediments downstream. Organisms living under these conditions are subject to different kinds and concentrations of radioactive materials. Bottom-dwelling animals such as *Chironomus* (a midge) and *Tubifex* (a burrowing worm) live in direct contact with the radioactive muds, whereas fish and planktonic organisms are found in the overlying water.

Factors Affecting Accumulation of Radioisotopes by the Biota

The accumulation of radioactivity by organisms is dependent upon a number of biological factors. Any living organism requires for its growth and development a combination of elements. These elements usually occur in the environment in quite different proportions from those found within the organism. There are functional processes by which an organism either selects or discriminates against the various elements as they are incorporated into protoplasm. The accumulation of radionuclides by aquatic organisms may be associated with either uptake through the food chain or uptake directly from the water and bottom sediments. The total accumulation determines whether human consumption of the aquatic organism will be hazardous. The factors that affect accumulation will be considered here.

The specificity of organisms, organs, and tissues for elements used in the structure and metabolism of the species is one important factor. Elements may be either concentrated or diluted in passing through the food chain, depending upon the organisms involved. Marine plankton organisms have been chemically analyzed for 10 different elements.[50] A diversity of

chemical behavior was found among the plankton organisms, and it was concluded that: "Certainly the safest generalization promises to be that, for any given chemical element, there will eventually be found at least one plankton species capable of spectacularly concentrating it." The specificity of an organism for an element may be the result of a metabolic requirement of a particular structure, such as iodine in the thyroid or strontium and calcium uptake in bones.

The influence of the metabolic function leads to a second important consideration in radionuclide accumulation, i.e., the excretion of radioactive elements is dependent upon the metabolic turnover of the particular element within the organism. Phosphorus and strontium, which are deposited in bone, do not turn over as rapidly as when they are deposited in soft tissues. The biological half lives of P^{32} in the bone and in the liver of rainbow trout are 240 and 20 days, respectively.[51] These data were obtained in experiments that also showed that it takes longer for P^{32} to reach equilibrium in bone than in soft tissue.

There have been differences reported regarding the mode of uptake of radioactivity by aquatic organisms. Uptake may occur through ingestion of food, absorption through tissues, or adsorption on surfaces in contact with the medium, with the latter two methods of uptake being less likely than the first method in terrestrial habitats. There is, however, insufficient information available at present to permit an evaluation of the importance of the various modes of uptake by the isotopes and organisms being considered, and additional research is required. For example, the uptake of Cr^{51}, Cu^{64}, P^{32}, As^{76}, and rare earths through ingested food has been found to be of more importance than sorption for fish living in the Columbia River below the Hanford reactors.[44] On the other hand, the uptake of Sr^{90}-Y^{90} by rainbow trout occurs both through food ingestion and through sorption from the water.[52] In fact, the amount of Sr^{90}-Y^{90} taken up by sorption is 10 times that from ingested food. Trout that were held in contaminated water and fed contaminated food concentrated Sr^{90}-Y^{90} equivalent to the sum of concentrations which resulted in two other groups of trout, of which one group was held in contaminated water and fed uncontaminated food and the other group was held in uncontaminated water and fed contaminated food. The passage of Cs^{137} through tuna skin and the almost complete exclusion of Sr^{89} and Ru^{106} have been observed.[53] Thus it is apparent that the uptake mechanism varies with the species of organism and the radioisotope involved.

The different radionuclides present in aquatic habitats are concentrated to varying degrees by different species. Concentration factors found upon analysis of comparable groups of organisms from White Oak Lake and the Columbia River have been found[54] to vary from 850,000 times for P^{32} by filamentous algae to 50 times for Cu^{64} by fish. The authors[54] concluded that the concentration factors of radionuclides by many of the lower forms of plant and animal life were greater than those for the higher forms, such as vertebrates. In general, Na^{24}, Cu^{64}, and the rare earths were found to have concentration factors of 2000 times or less; whereas Fe^{59}, P^{32}, and Sr^{90}-Y^{90} had concentration factors of 10,000 times or more. (A recent opinion from Hanford suggests a concentration factor nearer 10 than 10,000 for Sr^{90}-Y^{90}.)

The specific activity of an isotope has been frequently ignored in uptake studies, particularly from natural environments. At high specific activities a greater uptake of the radioactive isotope by organisms would be expected as the metabolic needs for the element are met. A nonlinear relation for isotopic dilution has been found. At low and high concentrations of phosphorus, with tadpoles, snails, and aquatic weeds, dilution was less effective than at intermediate concentrations.[55] Dilution of P^{32} with stable phosphorus was effective in reducing the P^{32} content of carp bone over the range of concentrations investigated. If isotopic dilution effects should prove to be a general phenomenon, the release of radioactive wastes to streams bearing a high mineral content would be advisable.

The biological similarity of pairs of elements, such as strontium and calcium or cesium and potassium, leads to what has been called a "sparing effect." For example, radioactive strontium may be utilized in the metabolism of an organism; however, with several species of of fish, it was found that, by increasing the calcium concentration fivefold, the Sr^{90} uptake was decreased by one-fifth.[56] A slight sparing effect was exhibited with the uptake of Cs^{137} by algae at relatively high concentrations of stable potassium; however, stable cesium was much more effective.[57]

The physiological condition of fish has been related to the body burdens of various radioiso-

topes.[58,59] Greater quantities of radioactive materials are accumulated when fish are actively feeding. Lower levels of contamination are observed at low winter or high summer temperatures, when the fish are less active.

Consequences of Biological Accumulation of Radioisotopes

It has been pointed out[46] that a human can drink about 2 liters of water per day from a source that contains a radionuclide at the maximum permissible concentration without exceeding the allowable body burden. If the water contains fish that have concentrated an isotope by a factor of 1000, humans can eat only about 2 g of such fish per day without exceeding the permissible body burden. Concentration factors for cesium by fish of 5×10^3 to 10^4 have been reported; therefore it may be relatively easy for humans eating fish from such an environment to exceed the maximum permissible body burden of cesium. Cesium-137 is a radioisotope that should be regarded with caution because cesium concentration increases through the food chain. This concentration increase occurs because ingested cesium is nearly completely absorbed by the feeding animal, and the organism will come to equilibrium at a higher contamination level than its food base.[60] The behavior of cesium demonstrates the human hazard that exists if radioisotopes are concentrated through the food chain. (D. J. Nelson)

Maximum Credible Accident Exposures at Reactor Site Boundaries

The increased utilization of atomic energy as a power source imposes upon the reactor designer, the reactor operator, and the government agencies that regulate the use of atomic energy an ever increasing responsibility to provide adequate safeguards to the general public. The present procedures for obtaining a construction permit and, finally, an operating license,[61] as well as the public hearings[62] on these two steps, make it highly probable that reactor plants are designed, constructed, and operated in accordance with the best standards of engineering practice. Following the initial startup of a reactor, it is important to maintain these standards through continuing programs of inspection, testing, and continued evaluation of safety, as well as administrative control.

In considering the safety of a particular reactor at a given site, the reactor designer investigates the various credible accidents and then investigates the consequences of the maximum credible accident for both the occupants of the facility and the inhabitants of the surrounding public areas. In those cases in which the maximum credible accident does not result in the release of large amounts of activity, it is common to analyze the results of a "hypotehtical accident," and often the results of such an analysis are used to establish the design criteria for the containment of the reactor. The pertinent calculations take into consideration the unique conditions at the reactor site, in addition to the reactor design and method of operation. There are many uncertainties inherent in such studies because the initial accident analyses must be made before the reactor design has been completed and before the operating procedures have been defined. In most cases, very little operating experience for a similar reactor exists from which to derive the information required for an accurate analysis.

The major factors that must be considered in evaluating the public-exposure hazard from the maximum credible accident are discussed here, and the estimated public-exposure doses at the site boundaries of the power reactors which are presently operating or are in the process of being built in the United States are compared in Table V-6. The information presented in Table V-6 is based on the most recent hazards reports available to the reviewer (see Section II, references 24 to 43); however, because of the preliminary nature of the information in many of the reports (lack of information on this subject in some), as well as lack of detail relative to assumptions, the comparisons of Table V-6 may only be considered to be the reviewer's interpretation.

The data of Table V-6 indicate that public exposure at the site boundaries of the various reactors, if the maximum credible accidents occurred, would, in general, be due mainly to inhalation of the fission products iodine and strontium, which result in the thyroid and bone doses, respectively. Inasmuch as the various reactor designers use different bases for computing exposure doses, a direct comparison of the exposure values given in the various hazards

Table V-6 POTENTIAL EXPOSURE DOSES AT THE SITE BOUNDARIES OF VARIOUS NUCLEAR POWER PLANTS FOLLOWING THE MAXIMUM CREDIBLE ACCIDENT*†

Short name of plant	Thermal power,‡ Mw	Percentage of total activity released from reactor following max. credible accident			Percentage of activity released from reactor which is available for release from containment vessel or reactor building			Max. containment vessel or building leakage rate, %/day	Assumed height of activity release from containment building, meters
		Noble gases	Halogens	Other	Noble gases	Halogens	Other		
PWR	231, 3 loops [241, 4 loops]	100	100	1§	100	100	*	0.15	0
Indian Point	585	25* 100	25 100	6.25 10	100	100	100	0.1	113
Yankee	392, initial; 485, potential	20	20	20§	100	100	100	0.2	0
N.S. *Savannah*	69	100	100	10	100	100	100	0.2	27.5
Saxton	20, initial; 28, potential	50	25	0.5	100	100	100	0.2	0
SPWR	60	90	45	4.5	100	100	100	0.1	0
VBWR	50	100	100	30	100	100	100	1.0	10
Dresden	630	100	100	10	100	*	*	0.5	10
Elk River	58, initial; 116, potential	8.62	8.62	0.43	100	100	100	0.063	0
Pathfinder	203	100	50	1	100	100	100	1.0 [0.2]	0
Humboldt Bay	163, initial; 200, potential	100	25	0.3§ [15]	100*	50*	50*	0.01 (dry-well chamber); 100 (building)	*
BONUS	50	100	100	30	100	100	*	0.25	0
Big Rock Point	156, initial; 240, potential	100	25	2	100	*	*	0.5	10
Hallam	240	0*	0*	0*				43	0
Enrico Fermi	300								
EGCR	10, loops¶; 84, reactor	100	100	10	100	10*	10*	2.2	22.9
FWCNP	153	4	4	4	100	100	100	0.25	0
Peach Bottom	115	100 [14]‡‡	100 [7]‡‡	100§ [17]‡‡	100	100	20	0.05 [0.2]	0 [uniform over containment]
Piqua	46	10	10	0	100	100		1.0	0–20.6
CVTR	63	100	60	1	100	100	100	0.1	0

Table V-6 (Continued)

Table V-6 (Continued)

Short name of plant	Meteorological parameters during inversion conditions: u (wind speed), meters/sec	n (stability parameter)	C (diffusion coefficient)	Exposure time,** hr	Distance to site boundary, meters	Exposure dose at site boundary,†† rem: Thyroid	Bone	Direct	Submersion	α (site parameter) (see Eq. 4)
PWR	1.3	0.5	$C_y = 0.12$; $C_z = 0.05$; $C_x = 0.05$	12 (8)	518	6000 (1.76×10^4)	1.6 (319)		0.8 (28)	1.09×10^{-2}
Indian Point					500	*	*		*	
Yankee	1.22	0.25	$C_y = 0.25$; $C_z = 0.20$	8	305	* (572)	* (1035)		* (1.36)	7.38×10^{-4}
N.S. *Savannah*	2.24	0.5	$C_y = 0.19$; $C_z = 0.09$	8	2000	0.068 (0.056)	0.013 (0.010)		0.068 (0.035)	1.02×10^{-4}
Saxton	3	0.55	$C_y = 0.4$; $C_z = 0.05$	8	277	228 [1500] (267.2)	3.2 [8.5] (9.65)	8.5 [6.0]	0.85 [0.8] (0.80)	4.84×10^{-3}
SPWR	1	0.5	$C_y C_z = 0.016$	8	458	20 (620)	6.8 (112)		0.28 (1.85)	6.28×10^{-3}
VBWR	1	0.5	$C^2 = 0.004$	3 8	600	20 (7.2×10^3)	8 (3.9×10^3)	4	4 (14.7)	3.63×10^{-3}
Dresden	1	0.5	$C = 0.05$; $C^2 = 0.004$	4 8	805	2400 (4.57×10^4)	19 (8300)	~1.0	2.4 (108)	3.63×10^{-3}
Elk River	1	0.5	$C^2 = 0.002$; $C_z = 0.02$	8	402	28.3 (1.5×10^3)	0.42 (136)		0.2 (3.6)	6.25×10^{-2}
Pathfinder	1	0.5	0.05	8	676 [880]	5×10^4 (19×10^4)	500 (6.88×10^3)	12	(565)	2.36×10^{-2}
Humboldt Bay	1–2	0.5	$C^2 = 0.002$	8	214	[0.3]§§	[0.05]§§	[1.0]§§	[0.4]§§	
BONUS	0.86	0.5	$C^2 = 0.0064$	8	483	580 (8650)	0.26 (4700)	32	1.6 (20.5)	1.73×10^{-2}
Big Rock Point	1	0.5	$C = 0.05$; $C^2 = 0.004$	8	805	1110 (4.3×10^3)	14 (615)	0.3	0.9 (18.1)	3.63×10^{-3}
Hallam	1	0.5	$C_y = 0.1$; $C_z = 0.06$	8	354	*	*		*	2.5×10^{-2}
Enrico Fermi										
EGCR	1.5	0.35	$C^2 = 0.01$	8	305	8.1 (6.95)	3.8* (1.3)	3.4	0.24 (0.05)	7.9×10^{-5}
FWCNP	1	0.5	0.05	8	683	1060 (1370)	110 (2480)	2 0.6	(3.25)	2.24×10^{-2}
Peach Bottom	1	0.5	$C_y = 0.32$; $C_z = 0.18$	8	914	130 [0] (144)	57.5 [0] (52.1)	[0.01]	0.4 [0] (0.34)	6.25×10^{-4}
Piqua	1.34	0.55	$C_y = 0.40$; $C_z = 0.05$	8	229	6 (336)		1.2	1.0 (0.8)	1.83×10^{-3}
CVTR	1.35	0.55	$C_y = 0.4$; $C_z = 0.05$	8	805	213 (334)	9.7 (10.1)	0.3	0.15 (1.56)	2.22×10^{-3}

Table V-6 Footnotes

*Remarks pertinent to calculations:

PWR — Exposure doses given in hazards report are smaller because of differences in assumptions as to a variable leakage rate from the containment vessel (factor of 2) and time of operation (factor of 3 for dose to bone) and because of assumption of removal of strontium as steam condenses. Condensation of steam in the containment vessel produced reductions in the amounts of cesium, rubidium, barium, and strontium proportional to the steam condensing in a given interval of time.

Indian Point — The containment vessel is surrounded by a concrete vessel, and air flow is provided to purge the annulus between the two vessels. Any leakage of fission products from the containment vessel is directed to the stack. As a result of the height of release, the resulting doses at the site boundary are very small. The concrete vessel reduces the direct dose at the boundary to a small value. The larger activity releases were assumed for inhalation exposure, the smaller for the direct exposure.

Yankee — The exposure doses at the site boundary are not given in the hazards report.

N.S. *Savannah* — There is no clearly defined site boundary for this reactor. Exposures are therefore tabulated for the distance to the maximum exposure for the conditions assumed. Activity is released due to leakage from the containment vessel. The activity released is then directed to a stack and is filtered before discharge. A filter efficiency of 99.9% is assumed for all fission products except the noble gases.

Saxton — Inhalation dose given in report is in curies of iodine or strontium; has been converted by the reviewer to dose in rem.

SPWR — Exposure doses given in the hazards report are lower because of differences in assumptions as to time at power (factor of 0.6 for exposure dose to bone), wind variability (factor of 6 to 8), and the mixing adjacent to the containment vessel (factor of 3).

VBWR — Exposure doses given in the hazards report are lower because of differences in assumptions as to fission-product release as a function of time (factor of 4) and exposure time (factor of 2 to 3). Since values of height of release, C, and n are not specified in report, they were assumed by reviewer based on similar analyses by the authors of the hazards report.

Dresden — Exposure doses given in the hazards report are lower because of differences in assumptions as to exposure time (factor of ~2), fuel meltdown variation with time (factor of ~2), variable leakage rate from containment vessel (factor of ~3), and washout by the spray system (factor of 2 to 5). Because of the action of the spray system, the amounts of strontium and iodine available for leakage from the containment vessel are reduced as a function of time. During the 8-hr period, effectively 50% of the iodine and 20% of the strontium are removed. Dose to the bone in the hazards report (19 r) is due only to strontium, other isotopes (mainly Ce^{144}-Pr^{144}) increase the dose to 40 r.

Elk River — Exposure doses given in the hazards report are lower because of differences in assumptions as to reduced leakage from the containment vessel, wind variability (factor of ~4), and breathing rate (factor of 2). Values indicated in the hazards report are reported as those for an 8-hr exposure but are actually for a 1-hr exposure.

Pathfinder — (No remarks.)

Humboldt Bay — Activity is released to the reactor building and, after passing through a cleanup system, is discharged from the stack at a controlled rate. The exposures at the site boundary are small. In postulating the amounts of fission products available for release, it is assumed that not more than 2% of the fission products released from the reactor core get to the suppression chamber. Only 10^{-4} of the fission products other than xenon and krypton are released from the suppression pool. Since the quantity of fission-product release from the suppression chamber is small compared with the release from the dry-well chamber, only the escape of fission products from the dry-well chamber is tabulated.

BONUS — The potential exposure dose to the thyroid was reduced in the hazards report calculations (factor of 20) by assuming that after a short period of time the spray system decreases the amount of iodine available for release to 1%. The dose to the bone is computed on the basis that only xenon, krypton, bromine, and iodine are released from the containment vessel.

BONUS — The potential exposure dose to the thyroid was reduced in the hazards report calculations (factor of 20) by assuming that after a short period of time the spray system decreases the amount of iodine available for release to 1%. The dose to the bone is computed on the basis that only xenon, krypton, bromine, and iodine are released from the containment vessel.

Big Rock Point — Exposure doses given in the hazards report are lower because of differences in assumptions as to fuel meltdown variation with time (factor of 2), variable leakage rate from the containment vessel (factor of ~3), washout by spray system (factor of ~5), and thyroid exposure dose (factor of ~2). The inhalation doses given in the report are in curies of iodine or strontium for 1- and 4-hr exposure and have been converted by the reviewer to dose in rem for an 8-hr period.

Hallam — In the hazards report calculation it is assumed that the maximum credible accident does not result in the release of large amounts of radioactivity, since it is postulated that the fuel elements do not fail. No comparison could be made by the reviewer because of insufficient information in the report regarding the assumptions made for the calculations. The inhalation of Na^{24} at the time of the maximum credible accident does not appear to be adequately covered in the report.

Enrico Fermi — The maximum credible accident for this reactor has not yet been defined.

EGCR — The original hazards report indicated that 10% of the iodine and strontium would be deposited. The supplement to the original report indicates that this factor of 10 may be obtained during an 8-hr period by use of a filter cleanup system inside the containment vessel instead of assuming the deposition. The dose to the bone is higher in the hazards report than that calculated by the reviewer because it was based on 2 years of operation.

FWCNP — Assumption of equal release of noble gases, iodine, and strontium is unrealistic.

Peach Bottom — Reactor is situated ~125 [225] ft from shore line of Susquehanna River. Calculations do not indicate dose rates at the shore line.

Piqua — Exposure doses are based on hypothetical core meltdown because the maximum credible accident does not release large amounts of activity. Numbers used are based on iodine not being retained in the coolant. The exposure doses given in the hazards report are lower because of a difference in the assumption as to fallout (factor of ~6.5).

CVTR — Inhalation doses given in report were expressed in curies of iodine or strontium and have been converted to dose in rem by the reviewer.

†Entries in brackets represent changes made by the reactor designer since the publication of the hazards reports used for this comparison. Since the reviewer has no additional information explaining such changes, these changes are only listed for additional information.

‡Unless otherwise noted, the hazard analysis is based on the higher power level listed.

§Amount of strontium assumed to be released.

¶Activity release at the time of the maximum credible accident is based on release of fission products from experimental fuel assemblies generating 10 Mw of thermal power in in-pile loops. Activity release from reactor core is assumed to be small by comparison.

**Exposures are computed as a function of time in most of the hazards reports. An 8-hr exposure time was chosen by the reviewer as a basis for comparison. Exposure times other than 8 hr are so indicated.

††Numbers in parentheses are values computed by the reviewer.

‡‡In the hazards report submitted to the AEC on July 25, 1960 (not reviewed in this article), the maximum credible accident results from failure of the fission-product purge line leading to the external trap. No concurrent pressure release occurs, and therefore it is assumed that no activity is released from the containment.

§§Maximum off-site dose for refueling accidents, much less for maximum credible operating accident.

reports and tabulated in Table V-6 cannot be used to indicate the relative safety of the various reactors. As will be discussed later, the many factors that enter into the calculations must be investigated in order to determine the degree of conservatism associated with a particular set of calculations. In order to obtain a comparison of the exposure doses to be expected at the site boundaries of the various reactors, the reviewer has computed the inhalation exposure dose from iodine and strontium and the submersion exposure dose from the noble gases and the halides. The resulting computed values are given in parentheses in Table V-6. A comparison of the doses computed by the reviewer and those listed as derived from the reports [some of the results in the reports were converted to an 8-hr exposure dose in roentgens equivalent man (rem) for purposes of comparison] indicates that, in general, the numbers listed in the hazards reports are substantially lower. The basic reason for the difference lies in the assumptions made for the two sets of calculations.

The doses computed by the reviewer are based on the following assumptions:

1. Submersion exposure is due only to krypton, xenon, bromine, and iodine.
2. Thyroid exposure is due only to radioactive iodine.
3. Bone exposure is due only to Sr^{89} and Sr^{90} (a better estimate may be made by including Ce^{144}).
4. The exposure time is 8 hr.
5. Activity release is instantaneous, i.e., the activity release from the fuel is not assumed to be a function of time after the maximum credible accident.
6. The leakage from the containment vessel is constant.
7. The breathing rate of a person at the site boundary is 500 cm^3/sec.
8. The reactor has operated at power for a period of one year.
9. The fission-product source has decreased only by radioactive decay.

Based on these assumptions, the submersion and inhalation exposure doses from iodine and strontium may be expressed as

$$E_I = (4 \times 10^{-2})\, P\beta f_I \alpha \qquad (1)$$

$$E_{Sr} = 1.81 \frac{f_{Sr}}{f_I} E_I \qquad (2)$$

$$E_\gamma^* = (1.75 \times 10^{-3})\, E_I \left(1 + 0.354 \frac{f_N}{f_I}\right) \qquad (3)$$

where E_I = iodine exposure dose, rem
E_{Sr} = strontium exposure dose, rem
E_γ = submersion exposure dose, rads
P = reactor power, watts
β = leakage rate from containment vessel, per cent per day
f_I, f_{Sr}, f_N = fraction of iodine, strontium, and noble gases released from reactor at time of maximum credible accident (instantaneous release)
α = site parameter describing the meteorological inversion condition

The site parameter α is defined by

$$\alpha = \frac{e^{-h^2/C_z^2 x^{2-n}}}{C_y C_z u x^{2-n}} \qquad (4)$$

where h = height of release, meters
C_y, C_z = diffusion coefficients in y and z planes
n = stability parameter
x = distance to site boundary, meters
u = wind speed, meters/sec

In order to compare the exposure doses obtained from Eqs. 1 to 4 with those given in the hazards reports, it is necessary to evaluate a number of factors which are unique to the given reactor design or which have been used by the reactor designer in his analyses. The main factors that lead to differences in the results are involved in assumptions 4 through 9, discussed previously, and include the following:

1. Exposure times given in Table V-6 vary from 4 to 12 hr.
2. Activity release in several reactors is assumed to be a function of time, as determined by the rate of fuel melting.
3. Leakage from the containment vessel is assumed to decrease with time as the pressure decreases.
4. The breathing rates assumed vary by a factor of as much as 2.

*For the case of an elevated release, the cloud gamma dosage is greater than that predicted by Eq. 3, out to distances comparable to the site boundary since, even though the cloud is aloft, the gamma irradiation can readily reach a receptor at the ground.

5. The assumed time of operation prior to the accident varies from less than one year to greater than one year. The time of operation influences only the calculation of the dose to the bone.

6. A decrease in the source of activity available for release as a result of radioactive decay was assumed in only about one-half of the exposure calculations. This difference influences the thyroid dose, but it has little effect on the dose to the bone.

7. Various mechanisms for reducing the amount of fission products available for release are assumed in the several reports, including deposition or plating out, washout by a spray system, cleanup with filters, and settling of activity inside the containment vessel as the steam condenses.

8. Various assumptions were made with respect to reductions in the amount of activity reaching the boundary as a result of either fallout or variations in wind direction.

The activity reductions expected from these various mechanisms for reduction are given in Table V-6 for particular reactors and in Table V-7 in more general terms. Inasmuch as it is somewhat difficult to separate the influence of one reduction factor from that of another, the factors of reduction cited are to be regarded as approximate. It is, of course, necessary to integrate all the reduction factors to obtain realistic estimates. The reviewer has felt it necessary, however, to evaluate the many factors individually when comparing the various reactors because there is no clear-cut consistent method used by reactor designers in estimating accident exposure doses at the boundary. The purpose of this review is not to indicate where factors of reduction are justified but to point out the magnitude of these factors and to indicate their real importance in evaluating the hazards of an accident. In many cases, the writers of the various hazards reports have not taken advantage of known justifiable reduction factors in order to be conservative. Thus the relative safety of the reactors listed in Table V-6 cannot be judged solely by either of the exposure values listed. It is necessary to evaluate many of the factors discussed below in reaching conclusions as to the accident exposures at any given site boundaries.

Table V-7 MECHANISMS FOR REDUCING ACTIVITY IN CONTAINMENT BUILDING

Mechanism	No. of reports that indicate activity reduction by listed mechanism	Factor of reduction of inhalation* exposure	
		Iodine	Strontium
Radioactive decay	9	1.04	1
Rainout (spray system)	3	2–20	5
Cleanup with filters	2	10–20	10–20
Deposition	1	2–5	2–5
Scrubbing	1	>1†	>1†
Settling	1	>1†	>1†

*Based on an 8-hr period; factors are only approximate.
†Magnitude of the factor is not evaluated here.

Maximum Credible Accident

For purposes of this review, the maximum credible accident is defined as the accident that results in the greatest exposure to the public at the site boundary. A comparison of the postulated maximum credible accidents for U. S. power reactors indicates that "the credibility of an assumed accident is subject to uncertainty in that operating experience is insufficient for accurate analyses, and the probability of a single or multiple failure is equally uncertain."[63] As a result of such uncertainties, a hypothetical incident is usually considered in instances where the maximum credible accident does not release large amounts of activity from the reactor. The difference between the maximum credible accident in one report and the hypothetical accident in another is frequently a question of terminology. In Table V-6, the exposure dose at the site boundary is for the maximum credible accident, except for those cases in which a hypothetical accident was also assumed.

Activity Release from the Reactor

Many of the factors that influence the amount of activity released from a reactor at the time of an assumed accident were discussed previously.[63] The total amount of activity released will depend on the fraction of the fission products released from the fuel that "melts" and the fraction of the total amount of fuel in the core that "melts." In addition to the determination of the total amount of activity release, it is important to know the amount of activity release as a function of time after the incident. In

general, there seems to be wide variation in the assumptions as to activity release from the reactor, as the comparisons in Table V-6 indicate. Much of the uncertainty as to what fraction of the fission products escapes at the time of the accident stems from the complete lack of information on meltdown of the fuel to be used in the proposed reactors.

In addition to the variations in the release values, the exposure doses given in the hazards reports are not comparable because some designers assume instantaneous release, whereas others consider release of the activity as a function of time (in six of the hazards reports, release is a function of time). The importance of the basis for comparison is emphasized by the fact that the assumption that the release is a function of time reduces the potential inhalation exposure by a factor of 1 to 4.

Furthermore, in computing the dose to the bone, it is necessary to estimate the amount of Sr^{90} released from the reactor. In several of the reports, it is indicated that the reactor has operated for an extended period and that there is equilibrium in the core. Although the fission products are generally at equilibrium, the Sr^{90} will not reach an equilibrium condition during the lifetime of the fuel elements. Therefore, for the comparison of the exposure doses at the site boundary, it was arbitrarily assumed in this review that the reactor had operated for one year at full power.

Method of Activity Removal

At the time of the maximum credible accident, if the fuel fails instantaneously, fission products will be transported into the containment vessel during the initial depressurization of the primary system. On the other hand, if the fuel does not fail instantaneously, the fission products that are released may escape from the core by diffusion or they may be carried off in the emergency coolant. There are several mechanisms for removal or reduction of the fission-product activity during the time between escape from the fuel and the time the fission products escape from the containment vessel. The importance of any of the mechanisms will depend upon the design conditions existing for the particular facility. As mentioned previously, the various mechanisms and the number of hazards reports that indicate activity reduction by a particular mechanism are given in Table V-7.

The mechanisms for reducing the activity available for leakage from the containment vessel exist to some degree in all reactor plants. Certainly, in all cases, credit can be taken for radioactive decay, and, if more information were available pertaining to deposition or particle size, the mechanisms of settling and deposition would be applicable to all reactors. Scrubbing, which may result from the action of the steam from the primary systems of water-cooled reactors (as well as from the steam systems of most reactors), will also be available for all reactors, although to different degrees. The mechanisms of rainout or filtration require the installation of a spray or filter system, respectively, in order to effect activity reduction. Since not all reactor plants are provided with these facilities, they cannot be counted on for reducing activity in all cases. From the number of reports in which credit is taken for the various mechanisms, it is obvious that much additional information pertaining to these mechanisms must be obtained.

Leakage from Containment Building

The comparisons of Table V-6 indicate that, in all cases except the Hallam Nuclear Power Facility and the Humboldt Bay Power Plant, pressure-shell containment vessels are provided to minimize the escape of fission products at the time of the maximum credible accident. The leakage rates from the containment vessels vary from 0.05 per cent for the Peach Bottom plant to 2.2 per cent for the EGCR. These leakage rates are determined primarily by establishing an allowable leakage that will keep the dose received external to the plant site at or below an allowable level. In all cases the leakage rates given are for the maximum allowable initial leakage from the containment vessel. The actual leakage will be determined by testing prior to operation. Additional information pertaining to the containment provisions for the various reactors is discussed in Section IV, pages 55 to 69.

The containment provisions for the Hallam and Humboldt Bay plants are somewhat different in that there is containment of the primary system in individual cells, as well as containment in the building. The high leakage rates from the buildings (43 per cent per day for Hallam and 100 per cent per day for Humboldt Bay) are representative of the worst conditions expected.

Leakage from the individual cells is at a very low rate.

In the hazards reports, there tend to be two methods of treating leakage from containment vessels: (1) the leakage is assumed to be at a constant rate for the period of interest, or (2) the leakage rate is assumed to decrease with time. In order to be able to compare the exposure doses at the site boundaries of the various reactors, the leakage rate was considered to remain constant with time. This assumption is not intended to imply that it is not proper to assume that the leakage rate decreases with time after the incident. On the contrary, the reviewer feels that it is justifiable to take advantage of a decrease in activity release if adequate information is available and that, in many cases, there tends to be too great conservatism with respect to methods of reducing the predicted hazards.

Two factors are of prime importance in reducing activity release from the containment vessel after the initial depressurization of the reactor primary system. The decrease in the temperature in the containment vessel following the accident will reduce the pressure rapidly. This decrease in temperature may be due to natural forces or it may be induced by special cooling systems, such as a spray system. In addition to the decrease in pressure, the reduced temperature will result in condensation of various isotopes, such as iodine, which are volatile at the temperature existing at the time of release from the primary system. In the case of the BONUS reactor, the potential dose to the thyroid is decreased by a factor of 20 by assuming that the leakage is decreased and that the fraction of iodine available for leakage is reduced. In six of the hazards analyses, it has been assumed that the leakage decreases with time and that the exposure dose at the site boundary is thereby reduced by a factor of 1 to 3.

In discussing the activity leakage from the containment building, it is necessary to know (1) the isotopic composition of the gas mixture that is released from the containment vessel and (2) how this composition compares with that of the gas mixture inside the containment vessel. In making hazards analyses, it is common practice to assume that the release of a given fission product is proportional to the volume leakage rate from the containment vessel. This assumption implies that the volatile and nonvolatile fission products are released at the same rate and, furthermore, that all nonvolatile fission products are released at the same rate regardless of particle size. Inasmuch as the leakage rate from a containment vessel is required to be small (2.2 per cent per day or less in the reactors proposed to date), it is expected that this leakage is either through the structural material or welds, or, in any event, through holes of very small diameter. If the leakage is predominantly diffusion through what might be considered open porosity in the materials or through small cracks, it would be expected that there would be preferential diffusion relative to the particle sizes of the nonvolatile constituents and to the molecular weights of the gases. Because of the lack of direct experimental information, the potential activity reduction by preferential leakage has been considered in only one hazards analysis, i.e., in the hazards analysis of the BONUS reactor it was assumed that only xenon, krypton, bromine, and iodine escape from the containment vessel.

Height of Activity Release

Of all the factors influencing the estimates of dose rates or integrated doses at site boundaries, perhaps one of the most significant is the assumption as to the height of the activity release. From the equations developed by Sutton[64] for expressing the dispersal of activity, the total integrated dose (TID) at some distance x from the source of release, assuming equilibrium atmosphere dispersion of a continuous source over a given period of time, may be expressed as

$$(\mathrm{TID})_h = \frac{B\, e^{-h^2/C_z^2 x^{2-n}}}{C_y C_z u x^{2-n}} \qquad (5)$$

where h = height of release, meters
x = distance downwind of the release, meters
n = stability parameter
C_y, C_z = diffusion coefficients for turbulence in y and z planes
B = constant that depends on the leakage rate of the various fission products, time, breathing rate, etc.
u = wind speed, meters/sec

If the release is at ground level, the total integrated dose may similarly be expressed as

$$(\mathrm{TID})_0 = \frac{B}{C_y C_z u x^{2-n}} \qquad (6)$$

The ratio of the total integrated dose for the case of elevated release to that obtained by assuming ground-level release, i.e.,

$$\frac{(\mathrm{TID})_h}{(\mathrm{TID})_0} = e^{-h^2/C_z^2 x^{2-n}} \qquad (7)$$

then gives the error factor of the exposure calculations if, in reality, the activity is released at ground level. On the other hand, if the exposure calculations are based on ground-level release and if it can be shown that the activity release is at some elevated point, a large safety factor is included in the results obtained.

In examining Eq. 7, it is apparent that the importance of evaluating the height of release increases as the distance to the site boundary decreases. (For the purposes of this discussion, the distance to the site boundary appears as x in Eq. 7.) In seven of the hazards reports for the reactors described in Table V-6, elevated release is assumed. At Indian Point and Humboldt Bay, activity discharge would occur from a stack, and thus the elevated-release assumption is certainly justified. In the other five analyses, elevated activity release or uniform release from the containment vessel, as at Piqua, were assumed. The assumption of elevated release introduces a reduction in the dose at the boundary which ranges from a factor of 3 for the Big Rock Point plant to a factor of ~75 for the Piqua reactor. It is apparent that the factors of reduction resulting from assuming elevated release may be so large that it is most important to evaluate this mechanism correctly. For those cases in which elevated release is assumed, it would be expected that, in addition to the normal testing of total leakage from the containment vessel, additional tests should be performed to prove that the leakage is not at ground level since any leakage at ground level is so much more significant than is elevated release in contributing to the dose rates at the boundary. In order to take credit for no leakage at ground level, it would appear that the designer should either provide special methods for preventing ground-level release (i.e., a barrier outside the first containment vessel, as in the case of the CETR) or be able to test the final containment structure for leakage at ground level and meet some specified leakage rate for distances from ground level to some height above ground level, h_0.

If the release from the containment vessel is assumed to be uniform with respect to its height, it may be shown that the dose rate, compared with that for ground-level release, is

$$\frac{D}{D_0} \approx \frac{1}{2h}\left(\pi C_y C_z x^{2-n}\right)^{1/2}$$

If the release is not uniform, the important factor to consider is the percentage of the leakage that occurs below h_0, where

$$h_0 \approx 1.5\left(C_y C_z x^{2-n}\right)^{1/2}$$

One final factor that must be considered when elevated release is assumed is the importance of the mixing of the atmosphere and the released activity as a result of air turbulence adjacent to the containment vessel. The turbulence may cause complete mixing, and the concentration of activity adjacent to the vessel may be expressed as

$$\chi = \frac{Q}{Au}$$

where χ = concentration of activity, curies/m^3
Q = source, curies/sec
A = cross-sectional area, m^2
u = wind velocity, meters/sec

However, before such mixing is assumed, it should be verified experimentally and compared with the effect of the mixing at small distances from the containment vessel. This may be done by computing the concentration from the expression

$$\chi = \frac{2Q}{\pi C_y C_z u (x + x_0)^{2-n}}$$

where

$$x_0 = \left(\frac{2A}{\pi C_y C_z}\right)^{1/(2-n)}$$

In the case of the Piqua reactor, where an elevated source was assumed, the above method was used.

Factors Reducing Activity During the Dispersal Period

During the time between the release of activity from the containment vessel and the time the activity reaches the site boundary, the initial amount of radioactivity at the time of release is reduced by radioactive decay and fallout. The importance of natural radioactive decay increases as the distance to the site boundary increases and the wind speed decreases, but, in general, it is an insignificant factor, except for extremely short-lived fission products. Chamberlain[65] has shown that the equations of Sutton may be modified by a factor to account for the removal of part of the activity by fallout. This factor may be expressed as

$$f_f = \exp\left(\frac{4v_g x^{n/2}}{nu\pi^{1/2}C_z}\right)$$

where v_g is the velocity of deposition and the other terms are as previously defined. For cases in which the settling velocity is much smaller than the horizontal wind speed, the settling velocity may be used for the velocity of deposition. In using this factor, the settling velocity will depend upon such factors as the element of interest, the particle size, electrostatic forces, impaction effects, thermal effects, and surface adsorption. Therefore, values of v_g should be obtained experimentally for conditions which stimulate the conditions typical of the maximum credible accident. This factor is only applicable for the case of a release at ground level. In the various hazards reports the computed exposure doses at the site boundaries do not, in most cases, take into account depletion of the released activity by fallout. The computation of thyroid exposure dose at the Piqua site boundary was based on activity depletion by fallout, and a factor of reduction of ~6.5 in the thyroid exposure dose resulted.

Meteorological Conditions at Site

The meteorological conditions at the site will have a large influence on the dispersal of the activity released from the containment vessel at the time of the maximum credible accident, and therefore, in order to compute the exposure dose at the site boundary and to make allowance for fallout, it is necessary to evaluate the diffusion coefficients C_y and C_z, the stability parameter n, and the wind speed u. Comparisons of the various hazards reports show that estimates of C, n, and u vary significantly from one site to another. Although some variation might be expected, in many cases the preliminary nature of the hazards reports suggests great uncertainty as to the accuracy of the values used for C, n, and u. Inasmuch as power reactors are normally constructed at remote locations and the meteorological conditions must be inferred from data taken at the nearest weather station, it would be surprising if the assumed meteorological conditions did represent the actual conditions at the site. It is therefore of interest to investigate the effect of changes in the meteorological conditions on the exposure dose at the boundary. The limits of the diffusion parameters used in the various hazards analyses are compared in Table V-8 with values established in USAEC Report WASH-740 (reference 66).

Table V-8 LIMITS OF DIFFUSION PARAMETERS FOR INVERSION CONDITIONS

Parameter	Values listed in hazards reports		Values listed in Report WASH-740	
	Maximum	Minimum	Maximum	Minimum
n	0.55	0.25	0.75	0.4
u	3	0.86	6	1
C_y	0.4	0.1	0.5	0.08
C_z	0.2	0.02	0.07	0.02
C^2	0.01	0.002		
C_yC_z	0.08	0.002	0.035	0.0016
C_z^2	0.04	0.0004	0.0049	0.0004

As shown in Table V-6, several of the hazards reports do not distinguish between C_y and C_z. For those cases in which ground-level release is assumed, it is implied that $C^2 = C_yC_z$; however, for those cases in which elevated release is assumed, if $C^2 = C_yC_z$, it would be inappropriate to use the value of C^2 rather than C_z^2. It must therefore be concluded that either $C_y = C_z$ or that there is an inconsistency in the application of the diffusion coefficients in those reports which do not differentiate between C_y and C_z. In Table V-8, values of C^2, C_yC_z, and C_z^2 are tabulated to indicate the variation in these three parameters.

The values in Table V-8 indicate the following effects on the dispersion calculations of the differences between the maximum and minimum values of u, C, and n:

Parameter	Range	Ratio of maximum value to minimum value
n	0.25–0.75	3
u	0.86–6	6.98
C_y	0.08–0.5	6.25
C_z	0.02–0.2	10

The distances to the site boundaries of the reactors listed in Table V-6 vary from 150 to 900 meters. Therefore the ratio of the maximum to minimum exposures, based on the diffusion parameter ranges for the "worst" inversion associated with the maximum credible accident, varies between 5.35×10^3 and 1.31×10^4 for the 150 to 900 meters, respectively. It should immediately be pointed out that such large variations are not real, because it is not usually possible to apply the minimum or maximum values to all the parameters, u, C_y, C_z, and n, at the same time. The uncertainty which results, however, from reasonable variations in any combination of u, C_y, C_z, and n or a large variation in a single parameter is quite significant. For example, the product C_yC_zu varies by a factor of about 30 for the reactors listed in Table V-6. Therefore, it can be concluded that until meteorological tests are performed at a reactor site, some uncertainty undoubtedly exists as to the amount of dilution of the activity released from the reactor during transit to the boundary.

In addition to the uncertainty in the accuracy of the diffusion parameters, there is the problem of estimating the exposure at the boundary for those short periods of time when the wind speed is essentially zero, i.e., a period of calm. During a period of calm, Sutton's equation does not apply. If a period of calm is represented by the complete lack of wind velocity, the concentration of activity adjacent to the reactor containment vessel will depend only on the molecular diffusivity of the activity, and the concentration of activity at distances greater than 100 ft and for times less than one day is negligible. It is clear therefore that the problem associated with the calm is not that created by a wind speed of zero, but, rather, that which would exist if the wind speed were greater than zero but less than 1 meter/sec during the period of interest (short times after the accident). It appears that data which give the wind velocity as a function of direction and time, with special emphasis on wind velocities of less than 1 meter/sec, should be obtained at the reactor site. The factor of wind variability has not, in general, been considered in the hazards reports. In the case of the Elk River Reactor, however, the calculated dose at the boundary was decreased by a factor of approximately 4 because of wind variability.

Site Boundary

Following the maximum credible accident, the exposure dose to the public will depend, of course, on the distance from the source of activity release to the surrounding uncontrolled public land. In many of the hazards reports, calculated dose rates or integrated exposure doses are given for various locations downwind of the activity release but not at the plant boundary. The various assumptions made for the hazards analyses raise questions as to what constitutes the site boundary and to what degree the operator of a reactor must exercise control over an area in order to consider it an exclusion area. In particular, what consideration must the operator give to penetrations of the exclusion area, such as highways, railroads, rivers, and lakes? In cases where highways or railroads are in close proximity to the reactor, it is conceded that proper control can be exercised at the time of the maximum credible accident to prevent the overexposure of individuals. However, the control procedures to be put into operation at the time of the maximum credible accident should be an integral part of the emergency operating procedures.

It is not quite so clear how the reactor operator can exercise control over rivers or lakes. Inasmuch as all power reactors depend upon a large heat sink, it is to be expected that the reactors presently being designed and the reactors of the future will be located on some large body of water. It is also probable that, in most cases, the distance from this body of water to the reactor will be small compared with the distance to the site boundary on land. It is common in the hazards reports to investigate only the exposure at the site boundary where the boundary is considered to be the minimum distance on land to where the plant property ends. Since the distance to the body of water in some cases may be less than 150 ft, the potential direct radiation and inhalation doses would be considerably higher at the adjacent body of water than at the locations considered. It would, of course, be possible to evacuate a person from a

lake or river adjacent to the plant site, but the dose rate during the initial period of exposure might be higher than that calculated for the site boundary. It would therefore seem to be necessary to analyze the exposure doses at locations near the reactor and to investigate both the time required to evacuate such locations and the degree to which these locations can be classified as restricted zones. In the case of a lake or a river, it might be possible to restrict the area adjacent to the plant. The real problem is not so much the exposure of traffic that would be in the critical zone for periods of less than 15 to 30 min but the exposure to individuals who use public bodies of water for fishing, pleasure boating, or swimming.

In the determination of the distance to the site boundary, attention should be given to surrounding "uninhabited areas." An adjacent area that is uninhabited at the time a reactor is built may not remain uninhabited if it is not within the control of the plant operator. It would therefore seem unjustified to consider the location of the maximum potential public exposure at any point beyond the actual plant boundary.

Exposure Time

In the determination of the potential exposure dose at a site boundary, it is necessary to assume some period of exposure. Two periods are of greatest interest: (1) a relatively short time that might be considered as an evacuation period and (2) a long time that would apply in cases where individuals would not be required to evacuate the area. In the selection of the exposure time, both the time an individual is exposed to irradiation and the time dependence of the activity release from the containment vessel are important. As an example, for the Big Rock Point plant, it is assumed that no fission products are released for 15 min after the maximum credible accident; the exposure dose is therefore small initially and does not reach a maximum until several hours after the initiation of the maximum credible accident. On the other hand, for the EGCR, the activity release has been assumed to be instantaneous, but the cleanup system provided inside the containment vessel is expected to reduce the activity available for release to negligible quantities in about 2 hr. Therefore, for the EGCR, the exposure dose received in the first 2 hr represents approximately the same as that which would be received over a long period of time. In order to provide a basis for comparison of the various reactors, an 8-hr exposure was arbitrarily assumed for the calculations of Table V-6.

Conclusions

The determination of the potential exposure dose at a reactor site boundary depends upon the assumptions made regarding the numerous factors discussed previously. In general, the hazards of a particular reactor have been judged by computing the dose rate, integrated dose, or concentration of activity at some location (not necessarily the site boundary). Since there are no official criteria with which to evaluate the hazard potential for reactor accidents, it is not surprising that the comparable 8-hr exposure values given in Table V-6 vary over a wide range. In addition to the large uncertainty associated with the assumptions and the large variation in the dose values, there is one other factor that casts doubt on the results presented in the hazards reports when the results are used to compare the relative safety of a particular reactor. Although the analyses define the exposure doses at certain locations and provide limited information on population densities, little information is generally provided concerning the integrated exposure doses to the general public. It would seem appropriate to include in the hazards reports calculations of the potential exposure doses to the public for the entire area that would be affected by an accident. In order to obtain an index of the relative hazard, it is important to know the maximum exposure dose rate for the public and the integrated exposure dose over a given area expressed in man-rem.

From a consideration of the various exposure calculations, it is apparent that criteria are needed relative to the conditions under which any single mechanism or combination of mechanisms may be employed in reducing the exposure dose at the site boundary. Several mechanisms involve only natural phenomena, such as radioactivity decay and fallout deposition, which may or may not be quantitatively defined. It may be presumed that a reduction mechanism can be used in an exposure calculation only if experimental data supporting its use for the case in question are available. Other reduction mechanisms involve physical processes such as spray cooling to minimize fuel-element melting or recirculation and filtration to reduce the activity

concentration in the gas leaking out of the containment vessel. In such instances the performance of these processes must be known under service (accident) conditions, and, before credit for these processes can be taken, it must be shown that they will operate in the event of any accident that might create the situation which the process in question was designed to ameliorate. (H. N. Culver)

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Section VI

CURRENT EVENTS

Holes in HRE-2 Core Tank

During two years of operation, between January 1958 and February 1960, there were several instances of overheating and severe local corrosion, melting, or burning of materials in the core of HRE-2, the second homogeneous reactor experimental assembly at the Oak Ridge National Laboratory. Two holes developed in the Zircaloy-2 core tank, the latest in January 1960. The perforated-plate flow diffusers were drastically attacked, as were corrosion specimens and a titanium holder, which were in the core during one run. These incidents occurred while the reactor was being operated with some uranium separated from the circulating fuel solution and with at least a part of the separated material in the core. Overheating from fissioning in deposits on the metal surfaces caused the drastic corrosion and burnout. There is evidence that such difficulties can be overcome by changes in the hydrodynamic design of the core and adjustment of the composition of the fuel. From the safety standpoint, experience with the HRE-2 assembly has (1) emphasized the importance of the relation between core design and properties of the fuel in fluid-fuel reactors, (2) confirmed that Zircaloy-water and titanium-water reactions involving massive pieces of metal are not normally self-sustaining, (3) provided further demonstration of the effectiveness of the negative temperature coefficient in controlling nuclear excursions in aqueous homogeneous reactors, and (4) emphasized the importance of knowledge of the distribution of fissionable material and careful control of inventory in fluid-fuel reactors.

The HRE-2 is a two-region circulating-fuel reactor designed to operate at 5 Mw(t) at temperatures up to 300°C and pressures up to 2000 psia. The reactor vessel assembly, including corrosion specimens in the core, is shown schematically in Fig. 6. A heavy-water solution of uranyl sulfate is recirculated upward through the core at a rate of about 450 gal/min. The design fuel solution composition is 0.04*M* UO_2SO_4, 0.02*M* $CuSO_4$, and 0.025*M* H_2SO_4. Copper catalyzes the recombination of radiolytic gas, and excess acid increases the stability of the fuel against separation of uranium by hydrolytic

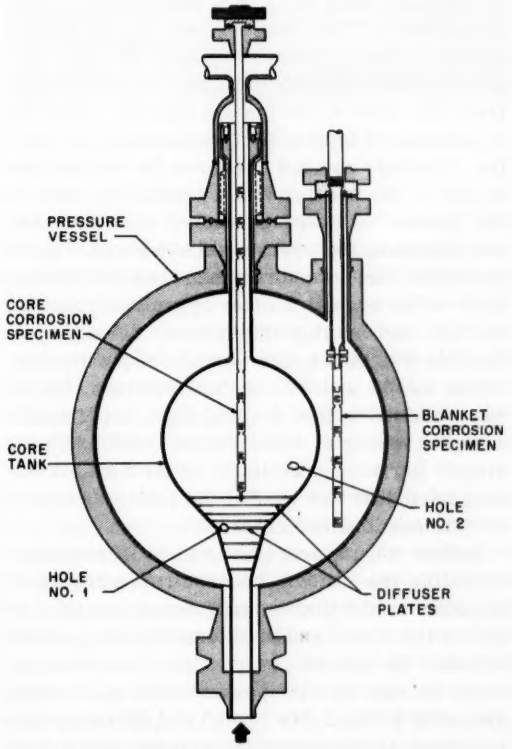

Figure 6—Diagram of HRE-2 reactor vessel.

processes. Initially the blanket contained only heavy water, which was recirculated at a rate of about 250 gal/min. The first hole[1] permitted communication between core and blanket, and the reactor has therefore been operated most of the time with uranium in both the core and blanket regions.

The formation of the first hole in the HRE-2 core occurred during run 14 on Apr. 4, 1958, about 30 min after the power had been raised to the nominal design level for the first time.[2] The actual heat-balance power level was 6.4 Mw. The reactor had been operated for a total of 989 hr, 257 hr critical, and approximately 400 Mw-hr since the first critical experiment. The initial indication of the incident was a decrease in the fuel level in the fuel circulating-system pressurizer. This was followed by changes in letdown-valve position, storage-tank weight, and other indications that there was transfer of fluid from the core to the blanket system. Analyses of samples of blanket fluid confirmed the transfer. The only obvious change in the nuclear behavior of the reactor was a gradual increase in the amount of power generated in the blanket, and therefore the power level was lowered by the operator. The reactor was subsequently shut down in the normal manner by gradually diluting the fuel and cooling the system. Evidence that the hole was in the core tank at the junction between the 30 and 90° conical sections (Fig. 6) was obtained within several days, but it was in a region where it could not be viewed with the simple periscopes available at the time. It was suspected that the weld at the junction between conical sections had cracked.

Before run 15, procedures were developed for operating the reactor with the concentration of uranium in the blanket only about one-third of that in the core,[3] making it possible to generate between 50 and 60 per cent of the heat in the core. In run 16 the power level was raised gradually to the 5-Mw level,[4] and operation was continued at this power for several days before the reactor was shut down for inspection and installation of corrosion specimens in the core. Brief surges or "pips" in power were observed during run 16, and there was one pip in which the power increased from 3.8 to 5.6 Mw. The disturbances caused no significant upset in the reactor temperature or pressure. Although tests were run in an attempt to characterize the disturbances, no conclusions were reached as to their cause. The total operating time at the end of run 16 was 2049 hr, 941 hr critical, and 2055 Mw-hr.

Following further unsuccessful attempts to view the hole, corrosion specimens were installed in the core, and run 17 was made.[5] Much of the power operation was at a level of approximately 3.5 Mw. Power pips occurred much more frequently than in run 16, and in one pip the power increased rapidly from 3.5 to 25 Mw as a result of the introduction of an approximately 2.1 per cent change in reactivity in 10 sec. The resulting pressure surge was about 200 psi. A pressure rise of 1000 psi, which would accompany the introduction of a 10 per cent change in reactivity in about 3 sec, is the maximum considered to be permissible during experiments with the reactor,[6] and a pressure rise of 3900 psi is considered to be dangerous.[7]

Two of three thermocouples installed in the corrosion specimen holder, one near the bottom and the other near the center, indicated rapid changes from the normal temperature to a temperature in excess of 1000°C at times not associated with power pips, and then they failed. Tests made at the time suggested that the abnormal behavior was due to sheath failure. The third thermocouple, which was near the core outlet, indicated normal temperatures throughout the run.

Correlation of analyses of samples of fuel solution showed that uranium separated from the fuel solution as the power increased and that the amount separated was approximately proportional to the power level, at least in the range of 1 to 3 Mw. The maximum power-dependent loss of uranium was about 1.5 kg. Comparison of the inventory based on analysis of the fuel and the inventory based on a relation between fuel concentration and critical temperature indicated that at least a part of the separated uranium was in the core and was effective in promoting the nuclear reaction. Run 17 was terminated after about six weeks for inspection of the core. The total operating time had been increased to 3134 hr, 1874 hr critical, and 4481 Mw-hr.

During the inspection it was found that the specimens in the core had come loose from the holder and had fallen onto the screens. The ends of the specimens were at least partially destroyed. Attachments to the holder were destroyed, and there were two holes in the webbing of the titanium holder. The two thermocouples that failed had disappeared up to the level of the

third thermocouple. The third thermocouple was intact, and specimens in the outlet pipe were intact. Many ligaments were missing from the diffuser plates, and it was discovered later that four of the top five screens had become detached from the wall.

The hole was seen for the first time from outside the core by means of a special arrangement of periscope and mirrors and, from inside the core, after a 2-in.-diameter hole was cut through the diffuser plates.[5] The hole is approximately 1 in. in diameter and gives the impression of having been formed by melting of the Zircaloy wall, which was $\frac{3}{8}$ in. thick at that point. Metallographic examinations of parts of the specimens and specimen holder[8,9] and of a small region of a section removed from the upper screen indicated that the metal in areas of severe damage had been heated to a very high temperature, presumably to the melting point. The heat-affected zones were so small and the metallographic structures were of such a nature that the heating and cooling must have occurred in very short times. Presumably the reactions ceased when the driving heat sources were removed as the metal disintegrated.

The remainder of the reactor operation, runs 18 through 21, was concerned with the investigation of effects of variables on the power level at which an instability is first detectable.[8,10–13] Otherwise unexplained increases or decreases in the reactor average temperature and the appearance of power pips were taken as indications of instability. These were ordinarily confirmed by analyses of fuel samples. The power level for inception of instability was found to increase with decreasing pressure and with increasing acidity of the fuel. The reactor operated in a stable manner at 5 Mw and 260°C if the pressure was kept below 1250 psia.

During the experiments the reactor was operated continuously at various power levels for periods of 2 to $3\frac{1}{2}$ months. The accumulated operating time was increased to 9179 hr, 6470 hr critical, and 12,187 Mw-hr before a second hole formed in the core tank[13] while borderline stability conditions were being investigated prior to raising the power to above 5 Mw. The second hole, measuring $1\frac{5}{8}$ in. long and $\frac{5}{8}$ in. across, and the end of a probing tool are shown in Fig. 7. Until the brittle edges were chipped away by the probing tool, the hole was smaller and irregular. This observation led to the possible, but less likely, conclusion that the second hole was

Figure 7—Second hole in HRE-2 core tank.

caused by corrosion rather than by burnout. The only indication of the incident was a gradual decrease in the average nuclear temperature and an increase in the fraction of power produced in the blanket as the mixing increased. The increased mixing made it necessary to operate with about equal concentrations of fuel in core and blanket, with most of the power produced in the blanket; therefore the experiments were terminated.

Results of the HRE-2 experience have led to the conclusion that difficulties were encountered because of a combination of factors related to the chemical stability of the fuel and the hydrodynamic design of the core. Uranium can separate from uranyl sulfate fuel solutions in several known ways. Solubility of the fuel constituents decreases with increasing temperature, and $UO_3 \cdot D_2O$, $UO_3 \cdot UO_2SO_4$, and $CuO \cdot 3UO_3$ can precipitate, or a heavy liquid phase that is rich in uranyl sulfate and copper sulfate can separate from solution if the temperature gets sufficiently above the design operating temperature. Uranium is sorbed by corrosion-product oxides of zirconium and iron, and there is some indication that the temperature coefficient of sorption is positive. Fuel constituents can be concentrated by boiling on the metal surfaces. Normally the concentration factor would be small, but severalfold increases in concentra-

tion might be obtained under unfavorable geometric and low flow conditions. Such concentrated fuel is much more corrosive to Zircaloy-2 than the normal fuel, and the corrosion products have been found to contain 10 to 40 per cent bound uranium in some experiments. The two-liquid-phase temperature of the fuel solution decreases with increasing concentration, passes through a minimum, and then rises again. This minimum two-liquid-phase temperature was below the boiling temperature of the fuel at the operating pressure during much of the operation.

The flow in the HRE-2 is sufficiently low that fuel in contact with some surfaces in a clean reactor might reach the boiling point. This condition is greatly aggravated when corrosion-product solids are present. Upward flow with low velocity tends to cause solids to accumulate on surfaces below the equator of the core, and thus to further retard the flow, increase the heating, and promote conditions that lead to fuel instability. Deposits that contain 40 mg of uranium per square centimeter should produce heat at burnout rates.[14] The fact that the flow conditions were unfavorable in the presence of solids was demonstrated in mockup tests[15] and was at least partially confirmed by examination of the reactor core.

Damage is confined to the lower half of the core tank; there is no evidence of similar damage to the upper hemisphere. New core tank designs that provide for high-velocity circulation of cool fuel over the metal surfaces and rapid removal of solids are being developed.[13] The effectiveness of much more modest changes in hydrodynamics, which can be obtained by removing the diffuser plates and reversing the flow in the present core tank, was inferred from model tests.[15] These changes are being made, and the holes are being patched so that the relation between solution stability and hydrodynamics can be investigated in the HRE-2.

Zirconium-water reactions are of considerable concern to the designers of water-cooled reactors with zirconium alloy parts. In the HRE-2, metal that was heated to the melting point by uranium accumulations was completely oxidized or dispersed, but the reaction did not propagate to adjacent metal. Removal of the diffuser plates involved cutting them into strips with a torch under water. Many small droplets of molten Zircaloy were formed during testing of the method and during the cutting in the core, but there was no evidence of a significant metal-water reaction. This experience seems to confirm other findings that zirconium pieces ignited in oxygen are normally quenched by water.[16,17]

Damage to materials in the HRE-2 core is of considerably less importance than the demonstration that there are ways in which uranium can separate from the fuel and deposit in the reactor system. The large negative temperature coefficient was shown to be effective in limiting the effects of shifting of this material in the reactor to pressure surges which were of little consequence. The amount of reactivity that can be controlled by the negative temperature coefficient is limited, however; and the ratio of uranium in the circulating system to uranium in the core of a large circulating-fuel power breeder reactor can be between 2 and 4. A major consideration, therefore, in the design of a large plant will be control of the inventory so that separation of fissionable material under abnormal conditions can be recognized and remedial measures can be taken before the amount of reactivity associated with the separated uranium is greater than that which can be controlled easily by the negative temperature coefficient.

(R. B. Briggs)

Shippingport Operating Experience

The Shippingport Atomic Power Plant, better known as the PWR, is a pressurized-water reactor system designed to operate at a thermal power level of 230 Mw. The reactor achieved initial criticality on Dec. 2, 1957, and first reached full power, i.e., 60 Mw net electrical output, on Dec. 23, 1957. The plant then operated more or less continuously for almost two years with the initial fuel loading, Core 1, Seed 1. The end of full-power operation with Seed 1 was reached at 5532.8 EFPH (equivalent full-power hours) on Sept. 6, 1959, at which time all 32 control rods were fully withdrawn. The plant was later operated for a short time at reduced temperature and power, bringing the total to 5806.1 EFPH, and was then shut down for refueling on Oct. 7, 1959. Following a period of testing, refueling, and minor modification, plant operation was resumed with Core 1, Seed 2, criticality being attained on Apr. 12, 1960.

Inasmuch as the PWR was the first large-scale nuclear power plant to be operated in the United States, it represents an unparalleled source of operating experience, much of which

bears directly or indirectly on nuclear safety.[18–36] Fortunately, the operating program has included extensive testing, and a determined effort has been made to document and make available the information and experience acquired. An excellent summary of the first year of operation was issued,[18] and the more recent experience is described in monthly progress reports of the Duquesne Light Company,[19–24] Bettis technical progress reports,[25–29] and reports of individual tests by the Duquesne Light Company (references 30, 31, and 33 to 36). These reports, in general, contain useful technical detail, although the scattering of the data inherent in this type of reporting is likely to discourage the casual investigator. Further summaries similar to reference 18 would be valuable and are presumably forthcoming.

Fission-Product Activity in Coolant

One area of continual interest is the level of fission-product activity in the primary coolant, since this is a measure of the integrity of the blanket elements, the first large-scale application of Zircaloy-clad UO_2-pellet fuel rods. In the PWR a fuel-element failure can be detected by (1) the Failed Element Detection and Location (FEDAL) system[37,38] and (2) radiochemical determination of the steady-state and transient distribution of fission-product activity in the coolant.

Radiochemical analyses during the early operating period indicated the presence of fission products in the coolant. Steady-state activity levels were established first during the period from April 28 to May 3, 1958, and again during the first 1000-hr full-power run from July 1 to Aug. 14, 1958. The levels had not changed significantly between the two periods. Subsequently, however, the activity levels gradually increased until April 1959, and then they remained substantially constant for the duration of Seed 1 life. Typical values are listed in Table VI-1.

Data available after one year of operation were evaluated[18] in an attempt to establish the source of fission products entering the coolant. The two most probable sources were uranium contamination of the surface of the Zircaloy and defective UO_2 blanket elements, since the observed activities were too high to be accounted for on the basis of the expected uranium content (1.35 ppm) in bulk Zircaloy. No direct evidence of surface contamination was found, although the apparently high delayed-neutron activity indicated by the FEDAL system was not inconsistent with this possibility. The most conclusive evidence of defective UO_2 elements was the peaking of I^{131} and I^{133} specific activities following startup (waterlogging effect). Also, the time required for ionic species to reach steady state in the coolant was different from that expected from a recoil source. It was concluded that the blanket contained one or more defective UO_2 rods, but it was not possible to determine the number of defective elements or whether significant levels of surface contamination were present.

Table VI-1 AVERAGE STEADY-STATE FISSION-PRODUCT ACTIVITIES IN PWR COOLANT

Isotope	Average specific activity, dis/(min)(ml)			
	April 1958*	July 1958*	April 1959†	August 1959‡
Cs^{138}	6,300	5,390	22,000	23,000
Kr^{88}	11,200	9,780	12,000	14,000
Br^{83}	167	168	950	1,000
Br^{84}	630	462	1,700	1,900
I^{131}	140	162	710	540
I^{133}	380	1,680	6,700	6,800

*See reference 18.
†See reference 20.
‡See reference 22.

The FEDAL monitor showed apparent peaking of delayed neutrons in the effluent from a blanket assembly (designated as assembly F2) following a reactor startup on Sept. 22, 1958, a characteristic indication of defective UO_2 rods.[30] In an attempt to determine whether failed blanket elements existed in other regions of the core, the FEDAL system was used to check nine other sampling positions that were most suspect.[31] No signs of bursts or peaks upon power increase during startup runs were observed which would indicate that elements had failed. The suspected blanket assembly and fuel bundles from two other blanket assemblies were replaced during the refueling and were then examined.[29] Results of the examination are not yet available.

In general, fission-product release experience at Shippingport indicates that both the seed and blanket elements have a high degree of integrity. The activity levels in the primary coolant are not yet approaching those for which the plant is designed.[32] It appears to the reviewers, however, that the value of the FEDAL

system has not yet been demonstrated. This system has required considerable attention and maintenance (references 19, 22, 25, 27, and 30), and the data collected have been largely inconclusive, with the exception of a strong indication that some defective UO_2 rods are present in blanket assembly F2. Unless the presence of defects is confirmed during the current examination of this assembly, the value of the FEDAL system will be in doubt.

Radiation Levels

Measurements of dose rates within the reactor plant container at full power have indicated that the intensities are much lower than were anticipated before plant operation. Selected values from a survey made after 1700 EFPH[18] are as follows:

Location	Radiation levels at full power, mr/hr	
	General	Maximum
Boiler chambers	100–700	14,000
Auxiliary chamber	0.02–2	400
Valve access cubicles	0.05–5	15
Isolated loop (48 hr after isolation)	0.1–4	20

It was possible for men to work in the auxiliary chamber and the purification-valve access cubicles during operation without limitation. Surveys (references 20 to 23, 33, and 34) have shown no significant increases in the dose levels during operation, except for a few points attributed to deposited corrosion-product activity (crud).

Radiation levels after reactor shutdown have not been a serious problem, but local levels from deposited crud increased sufficiently during the life of Seed 1 to be troublesome during refueling. Crud concentrations in the primary coolant remained essentially constant at about 3 ppb,[18,25–28] with occasional increases to as much as 60 ppb. The specific activity of the crud, after a 120-hr decay period, had gradually increased, however, to about 6×10^6 counts/(min)(mg) after 1700 EFPH[18] and to about 1.2×10^7 counts/(min)(mg) after about 5600 EFPH.[28] Later crud samples showed a higher percentage of long-lived activity, with Co^{60} predominating.

The maximum radiation intensity measured 15 hr after shutdown in the boiler chambers and in the auxiliary chamber was of the order of 10 mr/hr after 1700 EFPH.[18] At the end of Seed 1 life, a purification line in the access chamber showed 500 mr/hr, and a FEDAL pump base in the auxiliary chamber showed 180 mr/hr.[23] In a spot check of the reactor chamber 1 hr after shutdown at 1700 EFPH, the maximum intensity found was 150 mr/hr at the reactor head near a control-rod drive mechanism.[18] Later surveys of the same region revealed points at which the level was 1 r/hr after 4180 EFPH[20] and 1.3 r/hr after about 5600 EFPH.[23]

During refueling, radiation levels on the reactor head were as high as 1 r/hr just above the open mechanism ports and 25 to 75 mr/hr at the outside edge of the reactor head. An impeller removed from the 1D coolant pump showed 450 mr/hr at a distance of 2 in., and 1 r/hr was measured inside a handhole on the 1B heat exchanger. General levels, however, were less than 30 mr/hr.

The significance of the increasing shutdown dose levels is difficult to evaluate because the mechanisms of activity transport are not sufficiently well understood to permit extrapolation of existing data. Because the PWR core contains large quantities of Inconel and stainless steel at the edges, and there are relatively low coolant velocities through parts of the core, it was expected that the possibility of the development of sources of solid corrosion-product activity was potentially greater than in earlier pressurized-water reactor plants. Efforts to minimize these sources included (1) elimination of regions of low flow velocity wherever possible in the design and (2) operation of the primary coolant at *p*H 9.5 to 10.5, with LiOH as the additive.[18] Apparently, however, no attempt was made to reduce the normal cobalt content of core materials, as has recently been done in reactors with stainless-steel cores. Although the shutdown dose levels have not seriously impeded maintenance or the refueling operation, their troublesome nature is indicated by the experience during the first two weeks of the refueling operation, when 20 of the 60 maintenance men received permissible doses of 300 mr/week, and it was necessary to restrict their activities.[24] The concern of the present reactor staff for the potential seriousness of problems of this nature is perhaps indicated by the considerable attention being given to the development of decontamination procedures.[25–29]

Components

Many components in the PWR represent significant extrapolations in size from previously available equipment. Although extensive testing was carried out during development and after installation, a number of failures have occurred. It appears that these failures have not appreciably affected the safety or operability of the plant, but they perhaps give some insight into what to expect from well-developed components in an initial application.

Three main coolant pumps have failed. The incidents were spaced at intervals of several months, and, probably by coincidence, all three failures occurred in the same coolant loop. The first failure occurred on June 25, 1958, during a plant warmup with the pump operating at full speed. After the bearings were removed, examination indicated that they were in good condition and that the failure was caused by a defect in the stator can or that operation of the pump at a low system pressure had resulted in stator collapse.[18,26,35] The second failure, on Dec. 17, 1958, occurred when the plant was being cooled down and the pump was operating at slow speed. The bearings were found to be severely damaged. It was concluded that hydrogen, which came principally from common vent lines connecting the pressurizer and the pump, had caused steam and gas binding in the bearing area. The pressurizer vent was later separated from the common header.[21,29,39] The third failure occurred in October 1959, when the pump failed to start at slow speed after having been in a hot standby condition for about 25 days. The pump was removed for examination,[24,29] but results of the examination have not yet been reported.

One heat exchanger developed primary-to-secondary fluid leakage, and the steam drum sight glass failed on another heat exchanger. A hydrostatic test of one of the main heat exchangers in February 1959 showed two tubes to be leaking. Examination revealed that failure was caused by stress-corrosion, probably as a result of steam-blanketing and boiler-water chemistry deviating from the specification with respect to free hydroxide. Seventeen damaged tubes were plugged by expanding blind nipples inside the tube; two additional risers were added; and water-chemistry control was modified. About a year later, on Mar. 9, 1959, iodine activity was detected on the secondary side of the same boiler, indicating renewed leakage. The level of activity indicated[21] leakage at an estimated rate of about 0.2 ml/min. The iodine activity persisted, but the activity level [less than 1 dis/(min)(ml)] was not high enough to present a problem. At the end of Seed 1 life, a leak test showed nine tubes to be leaking, and the boiler was removed for examination.[18–20,24] In July 1959, the upper-level sight glass on another boiler shattered and allowed the boiler chamber to fill with steam. Attempts to enter the boiler chamber were unsuccessful because of the steam until the loop was shut down. A design change was initiated to provide combination ball-check and manual stop valves that would immediately isolate a defective sight glass (references 21, 22, 27, and 29).

Although all relief valves were tested under full-scale simulated plant conditions prior to installation, both of the self-actuated pressurizer steam-relief valves leaked during initial plant operation. Seat damage, apparently caused by foreign particles, was discovered and repaired, but slight leakage continued to occur whenever the system pressure approached the set point.[18] A measurement of plant leakage in September 1959 indicated[23] a leakage of 0.5 gal/hr from relief valves and a total primary plant leakage of 14.1 gal/hr (about 1.7 per cent per day).

Experience showed that there was a far greater amount of personnel traffic through the containment-vessel interlocks than had been anticipated during design. This indicated a need for a more positive indication of the closure status of airlock doors, as well as a more positive means for interlocking the inner and outer doors.[39] On one occasion, maintenance personnel attempted to open the inner door before the airlock test pressure was reduced to chamber pressure. The door blew open and was knocked out of alignment. A caution sign was placed on the outer door to prevent use until the damaged door was realigned.[21] Small submarine type access doors that are self-venting and have a mechanical interlock are to be mounted on the present doors of three airlocks, and larger pressure-equalization valves will be provided on the remaining airlocks.[25,29] Data[26] from a containment-vessel leak test at an internal pressure of 9.8 psig were extrapolated to predict a leaktightness of 0.1675 ± 0.0336 per cent of the net contained volume in 24 hr at the design internal pressure of 52.8 psig.

Kinetics

The seed-and-blanket type core of the PWR possesses the dynamic response and negative temperature coefficient characteristics of a small highly enriched core. The load-following capability was dramatically illustrated at the termination of the first 100-hr full-power run. Upon shutdown the turbine throttle was tripped with the plant at full power. The reactor was not on automatic control, and the reactor operator did not move rods manually until 20 to 30 sec after the throttle trip occurred. The automatically controlled steam dump was not in service. The resulting peak coolant temperature rise was 12°F, and the coolant pressure rise (180 psi) did not exceed the setting of the pressure-relief valves.[18]

Similar sudden-loss-of-load tests were performed frequently during Seed 1 lifetime. The reactor response continued to be very good, but, after a core life of about 2000 EFPH, throttle-trip tests indicated that either automatic rod insertion or controlled steam release was necessary to prevent operation of the pressurizer pressure-relief valve. The increased excursions in coolant conditions were attributed to a lowering of the magnitude of the negative temperature coefficient with core life.[25,26] The measured temperature coefficient gradually diminished from $-2.2 \times 10^{-4}/°F$ after 160 EFPH to $-1.0 \times 10^{-4}/°F$ at the end of Seed 1 life.[23,26] The plant[18] is expected to be stable as long as the negative temperature coefficient is as much as $-0.5 \times 10^{-4}/°F$.

The presence of extensive nuclear and thermal instrumentation in the PWR has enabled direct observation of xenon-induced instabilities. A successful attempt to override peak xenon in March 1959 resulted in the first observation that power oscillations were occurring across two opposite quadrants of the core. The oscillation phenomenon was subsequently studied during the next two 1000-hr continuous full-power runs. The oscillation was induced by intentional asymmetric displacement of control rods, followed by return of the control rods to the banked position. The period of oscillation was determined to be approximately 24 hr. The magnitude of asymmetry from the oscillations was limited to prescribed levels (± 6 per cent) by compensating asymmetric rod motion of the controlling groups of rods. In particular, PWR experience has demonstrated that, with adequate instrumentation, the reactor operator can be aware of such oscillations and that an informed operator can easily control their magnitude.[18]

Abnormal Operation

Operating statistics of the Shippingport plant are colored by the dual nature of the project objectives, i.e., power production and experimentation. The operating plan has included long full-power runs of approximately 1000-hr duration and shutdown periods of perhaps a month for physics measurements, plant tests, and training. Normal operations have been frequently interrupted to permit planned tests to be conducted. Such interruptions are in addition to those required for maintenance or by action of the safety system. On the other hand, certain design features that would probably not be included in a plant intended solely for power production, such as a complete spare primary coolant loop, have permitted operation at full power to continue while major maintenance was being performed. Thus, although it is perhaps interesting to note that the plant load factor (defined as the net power output divided by the theoretical net output if the plant operated continuously at full power) was less than 30 per cent for the period between criticality with Seed 1 and criticality with Seed 2 and that it was greater than 95 per cent during some extended power runs, such statistics have little or no significance as a measure of the capability of the PWR or other nuclear power plants.

As a routine matter, all scrams, safety shutdowns, and other significant incidents have been investigated and described in periodic progress reports.[19–29] A cumulative listing of such incidents during the first year of PWR operation is included in reference 18. The frequency of incidents has varied, of course, but a month of operation might typically include one or two safety shutdowns and four or five other incidents of appreciable import. A study of the incidents reported during the life of Seed 1 has revealed that none resulted in damage to the reactor or in a situation which threatened the uncontrolled release of appreciable amounts of radioactivity. Furthermore, there were no cases in which the safety system was called upon and failed to respond. Although the PWR safety-system experience is devoid of spectacular lessons, it is indicative of the situations encountered in the operation of a large nuclear power plant.

Loss-of-load accidents occurred on more than one occasion,[18] but the consequences were negligible because of the reactor response characteristics discussed previously. In one instance the station was back to 75 per cent power within less than 1 hr after the trip.[26] An example of the classical error of the operator turning the wrong handle resulted in a safety shutdown while Group III rods were being programed out to their limit of travel. Group IV rods were inadvertently moved instead of Group III, and, as a result, the nuclear level[27] exceeded the set point of 109 per cent.

An unusual incident involving the reactor containment vessel occurred when a false signal from the operational radiation-monitoring system resulted in automatic closure of the butterfly valves in the containment air-cooling system. Before the valves could be reopened, the air temperature and the pressure in the containment vessel increased to levels such that the containment pressure-trip device prevented opening of the butterfly valves. The pressure-trip device had to be disconnected while the cooling system was being returned to normal service.[20] On another occasion, while the reactor was critical, it was discovered that the "Test-Operate" switch was in the "Test" position rather than in the "Operate" position. In the "Test" position the maximum flux limit protection is removed from service, whereas in the "Operate" position all protection is in effect. When the shift engineer switched to "Operate," a safety shutdown occurred, which was later traced to a defective magnetic amplifier.[21] Apparently the design did not provide means for preventing or annunciating the disablement of a portion of the safety system in this manner.

A total loss of a-c power to the station resulted when one of the 138-kv-transformer lightning arresters was inadvertently sprayed with water during a routine fire drill. At the time, control rods were being withdrawn to attain criticality. A safety shutdown resulted, and the emergency transformer was placed into service.[19,25] This accident is significant only as an example of a highly improbable accident mechanism; it illustrates, however, why it is reasonable in an accident analysis to consider failure of vital portions of the plant even though no cause for failure is apparent.

The necessity for testing emergency procedures was illustrated during the performance of a simulated primary coolant leak test. In the test, control rods were bottomed, and a major rupture was simulated by valving a 40 gal/min leak in one of the loops. The operator was supposed to locate and isolate the leak by following an emergency procedure. Misinterpretation of the procedure resulted in the reactor being isolated from all four coolant loops. This error could have endangered the core by preventing normal circulation to remove decay heat and, if the rupture had occurred in the reactor vessel, by preventing escaping water from being replenished. The procedure was subsequently clarified to avoid this error.[36]

(H. C. McCurdy and T. D. Anderson)

Action on Reactor Projects by Licensing and Review Bodies

In the course of securing final approval for the operation of a power reactor, the applicant must clear several hurdles designed to assure the AEC, and thus the public, that the safety of the public has been adequately considered in a responsible manner. The procedure involves two major steps: (1) securing a construction permit and (2) securing an operating permit, as described in Title 10, *Code of Federal Regulations*.

Three proposed amendments affecting the licensing of production and utilization facilities were published in the *Federal Register*.[40] One amendment was intended to restrict the amount of work which may be done on a reactor facility before the issuance of a construction permit; specifically, it prohibits the laying of foundation and construction of any portion of the permanent facility on the site. The second proposed amendment provides for the issuance of a construction permit in instances where research and development on the reactor system may still be required. In such an instance a provisional permit will be issued, provided there is reasonable assurance from a safety standpoint that the site is suitable for a reactor of the power level and general design concept proposed. (The amendment was proposed before announcement of the court decision in the Power Reactor Development Company case, which is discussed at the end of this section, and it is not certain what effect this amendment, if it had been adopted earlier, might have had on that decision.) The third proposed amendment would permit the conversion of a construction permit into a pro-

visional operating license in advance of the scheduled startup so that transition could be made from construction to operation without delay. This would also facilitate the conversion from construction permit to operating license in cases in which involved features of the plant made it desirable to obtain actual operating experience before a full-term operating license was issued.

The processes involved in obtaining a construction permit and in obtaining an operating license require the preparation of a hazards summary report, which must be reviewed and approved by both the AEC Hazards Evaluation Branch (HEB) and the Advisory Committee on Reactor Safeguards (ACRS). As the final step before each license is issued, a public hearing is held. As a consequence of these various administrative reviews, any given reactor may be involved in reviews for a period of years. These reviews frequently result in modification of the reactor design or operation to enhance safety and, in any case, constitute the precedents for "adequate" safety.

Most power reactors proposed for construction and operation in this country are currently involved in some stage of the review process. The reactors discussed below have passed some milestone in the process during the quarter covered by this Review.

Big Rock Point Nuclear Power Plant

(AEC Docket 50-155)

A provisional construction permit has been issued to Consumers Power Company for the proposed boiling-water nuclear power station at Big Rock Point, Mich.[41] Issuance of the license followed approval by the ACRS and the HEB and a public hearing in Washington, D. C., on Mar. 29, 1960, in which the State of Michigan participated as an intervenor.

The reactor[42] will be a forced-circulation boiling-water plant having a maximum expected gross electric power generating capacity of 75 Mw [240 Mw(t)]. Although the construction permit covers a plant with this ultimate capacity, operation with the initial core is not expected to exceed 156 Mw(t). The steam generated in the reactor core will flow to the steam drum and then to a turbine, with no heat exchanger between. The reactor and its auxiliary equipment, except the turbine, will be situated within a containment vessel designed to adequately contain the radioactive materials released in the "maximum credible accident."

In concluding that the reactor of the proposed type and power level could be constructed at the proposed site with reasonable assurance that it could be operated without undue risk to the health and safety of the public, the ACRS commented[43] on the unusual $4\frac{1}{2}$-year research and development program and noted that the detailed design and experimental program will require additional study.

The intervention[44] of the State of Michigan, in behalf of the people of that state, resulted from the desire of Michigan authorities to assure themselves of the adequacy of the proposed waste-disposal practices and activity monitoring techniques. The testimony[45] of the HEB representative, J. J. DiNunno, noted that the hydrology of the site gave rise to safety concern but that the plant was to be designed to compensate for this feature by the use of retention basins, area and waste monitoring, and dikes for the control of surface runoff of accidental leakage.

The testimony of DiNunno further noted that the new features in the reactor include the use of fuel elements designed for a burnup of 15,000 Mwd/ton and provision for operation at core power densities of up to 45 watts/cm^3. The latter feature will be tested first on a partial core in the Vallecitos Boiling-Water Reactor, and such operation will not be attempted in the Big Rock Point reactor unless it has been proved that the high power densities can be achieved safely. Although it is expected that the core will be adequately instrumented for high-power-density operation, details of the instrumentation are not yet available. The HEB therefore reserved judgment on the instrumentation, as well as on the control system, which is to be similar to that in the Dresden power plant.

Among the other points raised at the hearing was the question of whether the proposed AEC regulation, which would prohibit the pouring of concrete for foundations before a construction permit was issued, would apply to the turbine building as well as to the nuclear portion of the plant.[45] Although both the Consumers Power Company and the AEC agreed that it did not, it was noted that there was some ambiguity in the proposed regulation as applied to single-cycle reactors. The AEC Hearing Examiner, in his intermediate decision,[41] which became final on May 28, 1960, ordered the issuance of a provisional construction permit and concurred in

the reservations expressed, as mentioned above, in the hearing testimony.

Boiling Nuclear Superheater Power Station (AEC Docket PP-4)

A hearing[46] on the granting of a construction permit to the Puerto Rico Water Resources Authority for the Boiling Nuclear Superheater (BONUS) Power Station at Punta Higüera, Puerto Rico, was held on Apr. 27, 1960. The reactor will employ a central forced-circulation boiling region and a peripheral four-pass steam-cooled region in which steam will be superheated and then passed directly to a turbine. The reactor[47] is designed for a thermal power level of 50 Mw, with an electrical output of 17.3 Mw. The complete reactor facility, including the reactor turbine-generator, fuel-handling facilities, and radioactive-waste-disposal facilities, is to be housed in a low-pressure hemispherical containment building of steel and concrete.

At its March 1960 meeting the ACRS reviewed the proposed plant and concluded that it could be constructed with reasonable assurance that it could be operated without undue risk to the health and safety of the public. The ACRS noted, however, that prior to operation there are several features which will have to be shown not to be hazardous. These include the radioactive contamination of the turbine, the long-term integrity of the fuel elements, and the nuclear superheater.

The AEC review of the facility was discussed by R. H. Bryan of the HEB.[48] In this case also, concern was expressed over the experimental nature of the superheater and the integrity of the fuel elements, particularly those in the superheater section. The concern with respect to the fuel elements was mitigated by the existence of provisions for detecting defective elements, the flexibility of operation with respect to power distribution, and the inherent protection of the primary system from damage from a superheater failure, as well as the use of reasonable hot-channel factors in estimating fuel-element temperatures. The HEB testimony also commented on the unique nature of the containment building and concluded that a structure of this type would provide adequate containment. This conclusion was reached despite the seemingly high site-boundary exposure doses calculated for the maximum credible accident, because the assumptions employed in making the calculations were "severely conservative." The only unusual site hazards were earthquakes and hurricanes, which, Bryan said, were adequately provided for in the design and location of the plant.

Carolinas-Virginia Tube Reactor (AEC Docket 50-144)

A provisional construction permit was authorized by the AEC Hearing Examiner for the Carolinas-Virginia Nuclear Power Associates pressure-tube reactor[49] in April 1960, and the decision became final on May 4, 1960. This action followed the hearing on Feb. 23, 1960, which was reviewed in the June 1960 issue of *Nuclear Safety*.[50] The Hearing Examiner, in submitting his intermediate decision, noted that the applicant had submitted amendments to his original proposal to the ACRS before the latter's final review of the proposal. Thus, although the ACRS did not specifically mention these amendments in its approval of the project, it must be assumed that they had the opportunity to do so. The construction permit was described as provisional in that an operating license will not be issued until evidence is presented on test results with respect to the novel features of the design, specifically, the use of heavy water, the U-shaped pressure tubes, and the Zircaloy-clad fuel elements.

Dresden Nuclear Power Station (AEC Docket 50-10)

The proceedings leading to the authorization of the Commonwealth Edison Company to operate the Dresden Nuclear Power Station at power levels up to 315 Mw (50 per cent of design) were discussed in the March 1960 issue of *Nuclear Safety*.[51] Fuel loading of the Dresden reactor for operation at power levels up to 315 Mw was halted, however, because of mechanical difficulties in the control-rod drive mechanism.[52] The trouble was investigated, and it proved to be due to the breakage of a shear pin which permitted the separation of certain sections of the control rods from their drive mechanisms. Modification of the rods to prevent this type of failure was reported by the General Electric Company, and subsequent tests of the drives involving over 500 scram cycles substantiated the reliability of the modified mechanism. As a consequence of this difficulty with the control-

rod drive mechanism, the AEC Division of Licensing and Regulation asked the Commonwealth Edison Company for additional information on the operation of the plant in the event a rod should become disengaged and the consequences which could be expected if a rod became stuck.

After the control rods had been modified, reinstalled, and tested, permission was granted for operation according to the previously approved test program at thermal power levels up to 315 Mw. The plant attained this power level on Apr. 13, 1960, and, in a subsequent report[53] submitted in accordance with their license agreement, it was concluded that "the reactor has exhibited extremely good characteristics of control and stability [and that there is] every indication ... that the reactor will continue to respond in a similar manner as higher operating powers are attained."

At its May 1960 meeting the ACRS reviewed the Dresden operational experience up to one-half design power and concluded[54] that there was nothing to "preclude continuing the approach to full-power operation under carefully planned test and operational procedures which will assure a prompt shutdown of the reactor in the event of significant departures from expected performance of the reactor or its control equipment." The ACRS went on to emphasize that:

> For some time to come ... operations at the Dresden power plant must be considered to be developmental in nature. The large size of the reactor and the relatively novel aspects of some of its control features go considerably beyond prior experience with boiling-water reactors .... Until such time as there has been a substantial period of operation at full power, the Committee advises that visitor access to the plant be restricted to those having an official reason for such visits. A hearing will be held to consider the presence or absence of good cause respecting the granting of an operating license for the full-power-level operation of the Dresden reactor.

Experimental Low-Temperature Process Heat Reactor Project

The Experimental Low-Temperature Process Heat Reactor (ELPHR) is a 40-Mw(t) pressurized-water reactor employing a single steam generator. The reactor is experimental only in that it will be the first reactor to be used to produce process steam rather than turbine-quality steam. The first site[55] proposed for this reactor was on Point Loma just outside San Diego Bay on the southern California coast. At its March 1960 meeting the ACRS reviewed[56] this site proposal and concluded that it would be unwise, from the safety point of view, to locate the reactor at this site at the present time. The reasons given were that the site had "unfavorable meteorology and high population density aggravated by recreational and fisheries aspects and lack of ocean dilution. The close proximity of the San Cabrillo Monument area [and] the experimental nature of the proposed installation" were factors. Objection to the site had also been voiced by the Navy, which owns adjoining property and has planned to set up a base for its nuclear-powered submarines.

After the Point Loma site was rejected, the AEC Division of Reactor Development, which is sponsoring the project, selected a site at Point Mugu, Calif., on the coast north of Los Angeles. The site report[57] was subsequently reviewed and approved by the ACRS. In a letter[58] that was issued following the May meeting of the ACRS, it was noted that:

> The proposed site appears to afford a degree of atmospheric dilution comparable with the most favorable that can be found in this coastal area. Because of the isolation afforded, as a result of site location relative to terrain features and government-owned land, the Committee believes that the indicated site is acceptable .... [However] it may be necessary to incorporate sufficient holdup capacity in the design of the plant to ensure that routine water-gas discharge to the atmosphere can be monitored and limited to periods of favorable atmospheric dilution conditions.

Hallam Nuclear Power Facility (AEC Docket PP-3)

The Consumers Public Power District proposes to construct and operate the Hallam Nuclear Power Facility (HNPF) in southeastern Nebraska. The plant is to consist of a 240-Mw(t) sodium-cooled graphite-moderated reactor (SGR) and the required associated equipment for generating electric power.[59] The reactor fuel is to be uranium-molybdenum alloy, and the plant is to be contained (not in a steel pressure-containment shell, as are most boiling- and pressurized-water reactor systems) in a building that is to be maintained at a subatmospheric pressure by continuous venting through filters.

In 1959, the ACRS reviewed the proposed facility and concluded[60] at that time that "consid-

ering the SRE experience, the isolated site ... and the design proposed ..., the proposed reactor can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public." The ACRS also noted that "the system is contained in a number of interconnected steel-lined concrete cavities [which should] be capable of containing fission products that might be released accidentally." However, "an improved filter system for collecting radioactive fumes from a sodium fire, should one occur, will be required."

Following the accidental melting of fuel elements in the SRE,[61] and in view of the fact that a portion of the ACRS safety evaluation of Hallam had been based upon the SRE experience, a report was submitted by the applicant on the applicability of the recent SRE experience to HNPF.[62] The report indicated that the SRE experience had been utilized to effect new design features that will be incorporated in the HNPF. The ACRS[60] then stated that it saw no reason to change its previous opinion relative to the construction of the HNPF.

On Apr. 8, 1960, a hearing[63] was held to consider the issuance of a construction permit. Testimony on health and safety aspects of the proposed facility was presented by M. B. Biles of the HEB. The site was characterized as favorable, as were salient features of the design, including low-pressure primary and secondary systems, multiple barriers between sodium and water, provisions to prevent and/or minimize the possibility of a sodium fire, the stability of the reactor, and the multiple safety devices. The dependence of the control of accidentally released activity on the operability of the filter was noted, and further developmental effort to preclude the possibility of filter plugging by sodium fumes was recommended. The hearing was routine, and the decision of the examiner will be announced at a later date.

Humboldt Bay Power Plant

(AEC Docket 50-135)

The Humboldt Bay Power Plant,[64] which the Pacific Gas and Electric Company proposes to construct on Humboldt Bay, south of Eureka, Calif., is to employ a 200-Mw(t) boiling-water reactor and pressure-suppression containment (see Section IV, p. 49). This latter feature, the most unique aspect of the proposed plant, is to consist of a concrete chamber, or dry well, in which the reactor is located and which is vented to a surrounding pool of water in another concrete vessel called the suppression chamber. In the event of an accident, the dry-well atmosphere would be vented through the pool water which would condense the steam released in the accident; consequently, only nominal pressures would result even from the maximum credible accident.

This reactor and its containment concept were reviewed by the ACRS at its March 1960 meeting. The ACRS concluded that the reactor system would be suitable for the proposed site if it were provided with the conventional pressure-vessel type of containment. The ACRS thought that the information provided did not make clear how much of the total reactor system would be housed within the proposed pressure-suppression chamber, nor did it "demonstrate the suitability of the steam-condensing system." Therefore the steam-condensing system cannot yet "be relied upon to protect the health and and safety of the public at this site."

The hearing on the granting of a construction permit, which had been set for Apr. 14, 1960, was subsequently postponed at the request of the Pacific Gas and Electric Company. The Company indicated that it would need more time to carry out further tests on the pressure-suppression containment system.

Prior to the ACRS decision, T. N. Saunders, Chief, Division of Industrial Safety, California State Department of Industrial Relations, notified the HEB of his objections to the proposed containment system, which were based upon (1) the fact that all the reactor system was not located within the containment system and (2) his belief that a pressure-vessel failure is as probable as a pressure-piping failure. This latter point has not been treated seriously in most hazards reports to date. The Pacific Gas and Electric Company justified the assumption that the pressure vessel would not fail on the reputed statistical fact that, in over 4000 boiler-years of operation at pressures in excess of 600 psia, there has never been a vessel failure.

Plum Brook Reactor Facility

(AEC Docket 50-30)

The National Aeronautics and Space Administration (NASA) has built a nuclear research reactor at the Plum Brook Research Facility near Sandusky, Ohio.[65] This 60-Mw(t) research reac-

tor is of the light-water-cooled and -moderated MTR type, with a primary beryllium reflector and secondary water reflector, and it is to be used in the development of a nuclear reactor system for aeronautical and space applications. NASA's request for an operating license for this facility is currently being reviewed. In February 1960, the Division of Licensing and Regulation specifically[66] noted that certain additional information should be submitted in order to complete the evaluation of the operating-license request. The specific information requested included (1) reactor operating principles and policies, (2) emergency procedures, (3) personnel qualification and training programs, and (4) a résumé of crucial parameters and numerical constants that constitute the specifications and limitations within which the plant will be operated. It is considered that the issuance of an operating license should be at least as dependent upon the quality of the operating program and personnel as on the integrity of the system design. Furthermore, in a test reactor it is essential to establish the limits beyond which the reactor may not be operated without an amendment to the operating license. The requested information was supplied by NASA in a supplement to their final hazards report.[67]

The ACRS reviewed the reactor facility at its March 1960 meeting[68] and concluded that the reactor "should be capable of being operated without undue hazard to the health and safety of the public." The ACRS specifically excluded comment on the operating procedures or the design of experiments, but they noted that in view "of the necessity of controlled release of radioactive gas and liquid effluents ... proposed experiments would have to be carefully reviewed and appropriate limitations [imposed]." It is presumed that the Division of Licensing and Regulation will require adequate assurances of the internal NASA review of experiments within the scope of the license and that operation or experiments beyond the specified scope will subsequently be reexamined as amendments to the proposed license.

Pathfinder Atomic Power Plant

(AEC Docket 50-130)

The Pathfinder Atomic Power Plant[69] is proposed for construction by the Northern States Power Company on the Big Sioux River, 3.5 miles northeast of Sioux Falls, S. Dak. The 203-Mw(t) reactor is to be a controlled-recirculation boiling-water system with nuclear superheat. The ACRS approval of this plant and the subsequent testimony and hearing on the issuance of a construction permit were discussed in the June 1960 issue of *Nuclear Safety*.[70] In an intermediate decision filed by the Hearing Examiner in April 1960, the issuance of a provisional construction permit was authorized.[71] The authorization was described as provisional because it was subject to the condition that:

> An authorization to operate the facility will not be issued by the Commission until the proposed research and development program has been completed [and satisfactory evidence obtained that] the novel reactor features respecting the fuel elements in the boiler region of the core, the incorporation of a superheater with unproven type of fuel elements as an integral part of the basic reactor core, and the proposal that a variable rate of primary coolant recirculation be utilized for control ... has been constructed and can be operated with adequate protection to, and without endangering the health and safety of, the public.

Peach Bottom Atomic Power Station

The Peach Bottom Atomic Power Station[72] consists of a 115-Mw(t) helium-cooled graphite-moderated reactor with a single steam generator located within the steel containment shell and a single turbine located outside the shell. The Philadelphia Electric Company proposes to construct and operate this plant at a site on the west shore of the Susquehanna River, 9 miles upstream of Conowingo Dam. The ACRS reviewed the proposal at its March 1960 meeting and concluded[73] that the site "provides a generally acceptable degree of isolation considered in relation to the proposed high integrity of containment." The ACRS had first noted that:

> The design of this reactor, although not yet fixed, will necessarily be such that routine reactor operation may be accompanied by considerable fission-product contamination of the coolant-gas stream [which places] particular emphasis on the need for reliability of the helium coolant system, the associated fission-product traps, and the outer containment shell .... The location of this reactor on Conowingo Pond, which is a potential supply of water to the City of Baltimore and serves several smaller cities, makes mandatory an especially careful consideration of factors which might lead to pond contamination.

Plutonium Recycle Test Reactor

The 70-Mw(t) Plutonium Recycle Test Reactor (PRTR),[74] now nearing completion at Hanford by the General Electric Company, is a vertical-pressure-tube reactor cooled and moderated with heavy water. The reactor is housed in a steel containment vessel, and heat is to be removed from the primary coolant by an intermediate light-water system and dumped into the Columbia River. The reactor is designed to obtain experimental data on plutonium fuel technology and plutonium fuel-cycle physics and to produce irradiated fuel for the development of plutonium fuel chemical reprocessing. At its January 1960 meeting the ACRS reviewed the PRTR and concluded[75] that "the proposed site and containment were suitable for the general reactor as conceived at that time." The ACRS noted, however, that "operating procedures and certain design details are still under consideration, and we must [therefore] reserve judgment until we are informed on the results of these studies." The results of the required additional studies were subsequently reported by the General Electric Company in a supplement to the original PRTR hazards report,[76] and, on the basis of the additional information, the ACRS concluded[77] at its May 1960 meeting that:

> Most of the details [previously mentioned by the Committee] are being satisfactorily settled. However, the following comments seem pertinent:
>
> 1. Filters should be installed at the outlet of the containment ventilating system.
>
> 2. Experimental information on void coefficients will be developed during the critical experiment program. Efforts to avoid a positive void coefficient should not be relaxed.
>
> 3. The Committee is skeptical of the claimed advantages of automatic control under any other condition than constant power.

Yankee Nuclear Power Station

(AEC Docket 50-29)

The Yankee Nuclear Power Station contains a pressurized-water reactor designed to operate at a power level of 485 Mw(t). The proposed operational plans for this reactor, as described in the Final Hazards Summary Report and Amendments 1 to 18, were reviewed by the ACRS, and approval of its operation was recommended, as mentioned in the June 1960 issue of *Nuclear Safety*.[78] A hearing was also held to consider evidence regarding the technical specifications for the reactor[79] since approval of the technical specifications was required in order to remove the provisional character of the construction permit, a necessary prerequisite to application for an operating license. The technical specifications were established, with the advice of the HEB, to identify the particular reactor being licensed, as well as to establish those systems or parts thereof which would affect the safety evaluation for that reactor to such an extent as to require AEC review and approval of any changes. Dr. C. K. Beck of the HEB testified that:

> With the exception of the details already discussed concerning the final design of the system for disposing of wastes from the secondary coolant loops, which we believe can be easily developed into a system that will be acceptable safetywise, we conclude that the integrated Yankee nuclear power plant design is satisfactory and that the Yankee Nuclear Power Station can be operated in accordance with the proposed procedures, schedules, and technical specifications, as revised at power levels up to 392 Mw(t), without undue risk to the health and safety of the public.

It should be noted that the approved power level is only 80 per cent of the design power level. Beck recommended that at least 500 hr of operating experience be accumulated at the 392-Mw level before a review is undertaken to ascertain the safety of operation at a higher level.

Subsequent opinions on the hearing were presented by both the Yankee Atomic Electric Company[80] (YAEC) and the AEC.[81] The YAEC counsel, D. G. Allen, commented on the propriety of approving at the construction stage the provisions in the Proposed technical specifications covering operating procedures and restrictions. In the proposed findings and conclusions submitted by the AEC counsel, Gerald Charnoff, it was recommended that the technical specifications be incorporated into the YAEC construction permit and that the latter be converted from a provisional to a definite construction permit. Commenting on the YAEC position with regard to safety reviews at an earlier stage, Charnoff stated that the Atomic Energy Act "allows for determination of the safety issues prior to the operating license proceeding to the extent permitted by the record, although such determination would be subject to review at the licensing proceeding in light of

new information or the showing of good cause as to why the license ought not be issued."

The Hearing Examiner, S. A. Jensch, in his intermediate decision[82] supported the YAEC position to the extent of deciding that the technical specifications should be conditionally approved and that final approval would have to await evidence on final design as reflected by completed construction. Jensch also ordered another hearing to consider issuance of an operating license.

In other actions on the Yankee Nuclear Power Station, the ACRS reviewed[83] Amendments 19 and 20. These two amendments concern the disposal of activity from steam-generator leakage and other design and procedural changes, including tests at 2000-hr intervals to determine the effects of plutonium buildup. The ACRS considered Amendments 19 and 20 to be satisfactory, but it stated that details of the test program will have to be approved when they become available.

Enrico Fermi Atomic Power Plant

(AEC Docket F-16)

The Enrico Fermi plant, which the Power Reactor Development Company (PRDC) proposes to construct at Lagoona Beach, Mich., includes a 300-Mw fast breeder reactor. The hearing on the issuance of a construction permit was reviewed in an earlier issue of *Nuclear Safety*,[84] and it was noted that the intervening AFL-CIO labor unions had filed suit in the U. S. Court of Appeals (AEC Docket 15,271) for the District of Columbia asking that the construction permit for this reactor be set aside. In a decision released in mid-June following the hearings in March,[85,86] the Court of Appeals held that there was insufficient evidence showing that the reactor could be operated safely, and therefore the court set aside the construction permit. The court thus upheld the unions' contention that:

> The AEC had violated its own rules in that it did not make certain findings in its August 1956 order issuing a provisional construction permit to PRDC for the fast breeder reactor at Lagoona Beach, Michigan.[87]

The unions held that the findings of the hearings were to establish whether:

> (1) There was reasonable assurance that the reactor may be constructed and operated at the proposed site without undue risk to the health and safety of the public.
>
> (2) The applicant is financially qualified to construct and operate the reactor for the period of the license requested, to pay [the] Commission for the nuclear material used during operation, and to use that material for a reasonable period of time.

In the brief by the attorney for the petitioners, it was claimed that "since the Commission did not make any of these findings, the permit to PRDC was issued in violation of the law and must be set aside." The attorney for the petitioners also contended that:

> It was the intention of Congress, in establishing a two-step licensing procedure, that an applicant who received a permit to construct a nuclear reactor should have substantial assurance that, if the reactor is built in accordance with the conditions of the permit, a license to operate will be granted, and therefore the essential findings with respect to safety should be made at the time the construction permit is issued. This protects the paramount interest of the public in safety, and, at the same time, encourages private industry to invest the very large sums of money required for such construction.

The brief filed by the AEC and the U. S. Government contended, however, that the Court of Appeals:

> ... has jurisdiction to review only a "final order" of the respondent AEC with regard to licenses or construction permits and only upon the petition of a "party aggrieved."[88] These statutory limitations require a "pre- and immediate" or "unavoidable threat" of injury.

It was pointed out also that the only source of grievance was the possibility that the PRDC reactor would operate in a hazardous manner. Since the AEC order authorized only construction and since a future hearing, following extensive investigation, would consider the operation of the reactor, it was inappropriate to anticipate what the ultimate action would be. The brief by the AEC further contended that:

> The terms of the Atomic Energy Act of 1954 establish a clear distinction between the safety considerations applicable to the two steps of the licensing procedure and authorize the Commission's policy. Section 185 directs the issuance of construction permits without prescribing specific standards, and vests in the Commission broad control over subsequent licensing of operation. Moreover, Section 182 explicitly conditions the licensing of operation upon the public health and safety requirements applicable

to reactors. Thus, the mandatory effect of the Act is to require definitive safety findings at the operating license stage. Reasonable safety standards for construction permits were left to the Commission's authority to make "such rules and regulations as may be necessary to carry out the purposes of this Act."

The U. S. Court of Appeals, in rendering its decision, held that the AEC was not justified, from a safety standpoint, in issuing a provisional construction permit for the plant in May 1959. If the decision holds, the AEC will have to suspend construction on the reactor until it can be shown that the project is unquestionably safe and that its location in a populous area is both necessary and not unnecessarily risky. The majority opinion held that, under the Atomic Energy Law, the AEC must make a finding as to the safety of the plant before issuing a construction permit. The opinion found the AEC arguments about the potential safety of the plant "ambiguous" and said that it had not been positively established that such a fast breeder reactor could be operated safely. The dissenting opinion objected to the apparent premise of the majority that, "because the Commission had permitted millions of dollars to be invested in construction of the plant, the AEC would feel bound to permit the plant to go into operation [regardless of safety considerations]."

(W. B. Cottrell)

Safeguards Reports and Selected Reading

The recently issued safeguards reports and selected literature pertaining to hazards and safety of reactors are listed below for reference. Because of the similarity of many reactors (in particular, the research reactors), the list is not intended to be all-inclusive.

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